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An Exelon/British Energy Company

10 CFR 50.90

September 30, 2002
5928-02-21096

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Three Mile Island, Unit 1
Operating License No. DPR-50
NRC Docket No. 50-289

Subject: License Amendment Request No. 318
Integrated Leak Rate Test Deferral

Dear Sir/Madam:

AmerGen Energy Company (AmerGen), LLC, hereby submits License Amendment Request No. 318, in accordance with 10 CFR 50.90, requesting an amendment to the Technical Specifications of Operating License No. DPR-50, for Three Mile Island, Unit 1. This proposed change will revise Technical Specifications (TS) Section 6.8.5 ("Reactor Building Leakage Rate Testing Program") to reflect a one-time deferral of the scheduled performance of the next Type A Containment Integrated Leak Rate Test (ILRT) from October 2003 to no later than September 2008.

In order to support the upcoming refueling outage at TMI, Unit 1, AmerGen requests approval of the proposed amendment by September 15, 2003 in order to avoid costs associated with pre-staging equipment and personnel associated with performing the ILRT.

Once approved, this amendment shall be implemented within 30 days of issuance.

Additionally, there are no commitments contained within this letter.

A017

A copy of this License Amendment Request, including the reasoned analysis about a no significant hazards consideration, is being provided to the appropriate Pennsylvania State official in accordance with the requirements of 10 CFR 50.91(b)(1).

If you have any questions or require additional information, please contact John Hufnagel at (610) 765-5507.

I declare under penalty of perjury that the foregoing is true and correct.

Respectfully,

09-30-02
Executed on

Michael P. Gallagher
Director, Licensing and Regulatory Affairs
Mid-Atlantic Regional Operating Group

Attachments: 1-Licensee's Evaluation
2-Markup of Technical Specification Pages
3-Final Technical Specification Changes
4-Risk Assessment for TMI Unit 1 to Support ILRT (Type A) Interval Extension Request

cc: H. J. Miller, Administrator, Region I, USNRC
USNRC Senior Resident Inspector, TMI
T. G. Colburn, USNRC Senior Project Manager
File No. 02079

ATTACHMENT 1

THREE MILE ISLAND
UNIT 1

Docket No. 50-289

License No. DPR-50

LICENSE AMENDMENT REQUEST NO. 318

"Integrated Leak Rate Test Deferral"

Supporting Information - 11 Pages

ATTACHMENT 1 CONTENTS

1.0 INTRODUCTION

AmerGen Energy Company (AmerGen), LLC, Licensee under Facility Operating License No. DPR-50 for Three Mile Island (TMI), Unit 1, requests that the Technical Specifications to the Operating License be amended to revise Technical Specification Section 6.8.5 ("Reactor Building Leakage Rate Testing Program") to reflect a one-time deferral of the scheduled performance of the next Type A Containment Integrated Leak Rate Test (ILRT) from October 2003 to no later than September 2008. The marked up Technical Specification page and final Technical Specification page are contained in Attachments 2 and 3, respectively. Attachment 4 contains the "Risk Assessment for TMI, Unit 1 to Support ILRT (Type A) Interval Extension Request."

2.0 DESCRIPTION OF THE PROPOSED AMENDMENT

The proposed change involves a one-time exception to the ten (10) year frequency of the performance-based leakage rate testing program for Type A tests as required by Nuclear Energy Institute (NEI) 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J."

The proposed change would revise Section 6.8.5 ("Reactor Building Leakage Rate Testing Program") of the TMI, Unit 1 Technical Specifications to add the following statement:

> "as modified by the following exception to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J":
>
> a. Section 9.2.3: The first Type A test performed after the September 1993 Type A test shall be performed no later than September 2008."

3.0- BACKGROUND

The proposed change involves a one-time exception to the ten (10) year frequency of the performance-based leakage rate testing program for Type A tests as required by Nuclear Energy Institute (NEI) 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J." The current ten (10) year Containment Integrated Leak Rate Test (ILRT) for Three Mile Island, Unit 1 is due in October 2003. The proposed exception would allow the next ILRT for TMI, Unit 1 to be performed within fifteen (15) years (September 2008) from the last ILRT as opposed to the current ten (10) year frequency.

This one-time exception will result in the following:

- Maintaining plant safety as demonstrated by past-ILRT history, probabilistic risk assessment, and ongoing containment inspections.

- Performing a Type A Containment ILRT during Refuel Outage T1R17, currently scheduled for October 2007.

- Cost savings have been estimated for the T1R15 outage at approximately $1.5 million, which includes labor, equipment and two days of critical path outage time needed to perform the test. Personnel radiation exposure reduction for T1R15 is estimated at .5 person-rem.

4.0 REGULATORY REQUIREMENTS & GUIDANCE

a. 10 CFR 50, Appendix J, Option B

The testing requirements of 10 CFR 50, Appendix J, provide assurance that leakage from the primary containment, including systems and components that penetrate the containment, does not exceed the allowable leakage values specified in Technical Specifications. The limitation on containment leakage provides assurance that the primary containment will perform its design function following plant design basis accidents.

10 CFR 50, Appendix J was revised, effective October 26, 1995, to allow licensees to perform containment leakage testing in accordance with the requirements of Option A, "Prescriptive Requirements," or Option B, "Performance-Based Requirements." Technical Specification Amendment 201 was issued to GPU Nuclear Corporation (dated May, 1997) to permit implementation of 10 CFR 50, Appendix J, Option B. Amendment 201 revised Technical Specification Section 6.8.5 to require Type A, B and C testing frequency in accordance with programmatic controls established to implement Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program." RG 1.163 specifies a method acceptable to the Nuclear Regulatory Commission (NRC) for complying with 10 CFR 50, Appendix J, Option B by approving the use of NEI 94-01 and ANSI/ANS 56.8-1994, subject to several regulatory positions in the guide.

Deviations to RG 1.163 are permitted by 10 CFR 50, Appendix J, Option B, as discussed in Section V.B, "Implementation." Therefore, this application does not require an exemption from 10 CFR 50, Appendix J, Option B.

Adoption of the Option B performance-based containment leakage rate testing program did not alter the basic method by which Appendix J leakage rate testing is performed; however, it did alter the frequency at which Type A, B and C containment leakage tests must be performed. Under the performance-based option of 10 CFR 50, Appendix J, test frequency is based upon an evaluation that reviews "as-found" and "as-left" leakage history to determine the frequency for leakage testing which provides assurance that leakage limits will be maintained.

The allowed frequency for Type A testing, as documented in NEI 94-01, is based, in part, upon a generic evaluation documented in NUREG-1493. The evaluation documented in NUREG-1493 included a study of the dependence of reactor accident consequences on containment leak-tightness for five reactor/containment types including Zion, Unit 1, a large, dry containment building. The TMI, Unit 1 containment is similar in design to the Zion containment. NUREG-1493 made the following observations with regard to decreasing the test frequency:

- Reducing the Type A Integrated Leak Rate Test (ILRT) testing frequency to one per twenty (20) years was found to lead to imperceptible increase in risk. The estimated increase in risk is small because ILRT's identify only a few potential leakage paths that cannot be identified by Type B and C testing, and the leaks that have been found by Type A tests have been only marginally above the existing guidelines. Given the insensitivity of risk to containment leakage rate, and the small fraction of leakage detected solely by Type A testing, increasing the interval between ILRT testing has minimal impact on public risk.

- While Type B and C tests identify the vast majority (greater than 95%) of all potential leakage paths, performance-based alternatives are feasible without significant risk impacts. Since leakage contributes less than 0.1 percent of overall risk under existing guidelines, the overall effect is very small.

NEI 94-01 requires that Type A testing be performed at least once per ten (10) years based upon an acceptable performance history. Acceptable performance history is defined as two consecutive periodic Type A tests at least 24 months apart where the calculated performance leakage rate meets acceptable limits. Based upon the acceptable January 1991 and September 1993 ILRTs, the current test interval for Three Mile Island, Unit 1 is once every ten (10) years, with the next test currently scheduled be performed by October 2003.

5.0 TECHNICAL ANALYSIS

a. 10 CFR 50, Appendix J, Option B

The testing requirements of 10 CFR 50, Appendix J, provide assurance that leakage from the primary containment, including systems and components that penetrate the containment, does not exceed the allowable leakage values specified in Technical Specifications. The limitation on containment leakage provides assurance that the primary containment will perform its design function following plant design basis accidents.

10 CFR 50, Appendix J was revised, effective October 26, 1995, to allow licensees to perform containment leakage testing in accordance with the requirements of Option A, "Prescriptive Requirements," or Option B, "Performance-Based Requirements." Technical Specification Amendment 201 was issued to GPU Nuclear Corporation (dated May, 1997) to permit implementation of 10 CFR 50, Appendix J, Option B. Amendment 201 revised Technical Specification Section 6.8.5 to require Type A, B and C testing frequency in accordance with programmatic controls established to implement Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program." RG 1.163 specifies a method acceptable to the Nuclear Regulatory Commission (NRC) for complying with 10 CFR 50, Appendix J, Option B by approving the use of NEI 94-01 and ANSI/ANS 56.8-1994, subject to several regulatory positions in the guide.

Deviations to RG 1.163 are permitted by 10 CFR 50, Appendix J, Option B, as discussed in Section V.B, "Implementation." Therefore, this application does not require an exemption from 10 CFR 50, Appendix J, Option B.

Adoption of the Option B performance-based containment leakage rate testing program did not alter the basic method by which Appendix J leakage rate testing is performed; however, it did alter the frequency at which Type A, B and C containment leakage tests must be performed. Under the performance-based option of 10 CFR 50, Appendix J, test frequency is based upon an evaluation that reviews "as-found" and "as-left" leakage history to determine the frequency for leakage testing which provides assurance that leakage limits will be maintained.

The allowed frequency for Type A testing, as documented in NEI 94-01, is based, in part, upon a generic evaluation documented in NUREG-1493. NUREG-1493 made the following observations with regard to decreasing the test frequency:

- Reducing the Type A Integrated Leak Rate Test (ILRT) testing frequency to one per twenty (20) years was found to lead to imperceptible increase in risk. The estimated increase in risk is small because ILRTs identify only a few potential leakage paths that cannot be identified by Type B and C testing, and the leaks that have been found by Type A tests have been only marginally above the existing guidelines. Given the insensitivity of risk to containment leakage rate, and the small fraction of leakage detected solely by Type A testing, increasing the interval between ILRT testing has minimal impact on public risk.

- While Type B and C tests identify the vast majority (greater than 95%) of all potential leakage paths, performance-based alternatives are feasible without significant risk impacts. Since leakage contributes less than 0.1 percent of overall risk under existing guidelines, the overall effect is very small.

NEI 94-01 requires that Type A testing be performed at least once per ten (10) years based upon an acceptable performance history. Acceptable performance history is defined as two consecutive periodic Type A tests at least 24 months apart where the calculated performance leakage rate meets acceptable limits. Based upon the acceptable January 1991 and September 1993 ILRTs, the current test interval for Three Mile Island, Unit 1 is once every ten (10) years, with the next test scheduled to be performed by October 2003.

b. TMI, Unit 1 Integrated Leak Rate Test History

Type A testing is performed to verify the integrity of the containment structure in its Loss of Coolant Accident (LOCA) configuration. Industry test experience has demonstrated that Type B and C testing detect a large percentage of containment leakages and that the percentage of containment leakages that are detected only by integrated containment leakage testing is very small.
Three Mile Island, Unit 1 has undergone seven operational Type A tests in addition to the pre-operational Type A test. The results of these tests demonstrate that the Three Mile Island, Unit 1 containment structure remains an essentially leak-tight barrier and represents minimal risk to increased leakage. These plant specific results support the conclusions of NUREG-1493. The Three Mile Island, Unit 1 ILRT results are provided below:

Test Date	Acceptable Limit Note 5	Leakage Rate Note 5
3/74 (Pre-Operational)	0.075	0.043
4/77	0.075	0.103
Retest (Note 1)	0.075	0.042
4/78	0.075	0.064
7/81	0.075	0.028
4/84	0.075	0.042
11/86 (Note 2)	0.075	0.1
Retest	0.075	0.034

Test Date	Acceptable Limit Note 5	Leakage Rate Note 5
12/90 (Note 3)	0.075	0.096
Retest	0.075	0.013
9/93 (Note 4)	0.075	0.072

Notes:

1. The initial leak rate testing performed between 10:00 on 4/16/77 and 16:00 on 4/18/77 was not successful. An extensive search failed to identify any significant sources of leakage, however a shift in the trend of the containment mass points occurred after 16:00 on 4/18/77. The cause of the first test failure was determined to be leakage into volumes internal to the containment building.

2. The initial ILRT testing revealed leakage past reactor building purge valve AH-V-1A and AH-V-1B interspace isolation valves PP-V-101 and PP-V-102. This leak path was eliminated and the test was successfully re-performed. Currently, this leakage path is local leak rate tested on a quarterly frequency.

3. The first test was declared invalid due to once-through steam generator valve leakage. These valves were out of their normal position and outside the test envelop. Valve lineup guidance was added to the procedure and these valves were shut or isolated as required. The ILRT was re-performed successfully. Subsequently, all Hancock 5500 W instrument root and drain/vent skin valves on the once-through steam generators were replaced with a different design valve which is much less prone to body-to-bonnet flange leakage.

4. During the 1993, TMI-1 performed a combination ILRT/LLRT "as-found" test at the beginning of the refueling outage, which represented a different method from that used in the past. Approximately two-thirds of the LLRTs were performed just prior to the "as-found" ILRT during the stabilization period. Also, during the ILRT larger-than-normal variations in reactor building temperature were observed. These variations, while within acceptable bands, may have also influenced the test results.

5. Leakage rates are expressed in units of containment air weight percent per day at test pressure (50.6 PSIG). Calculated results are based on the mass point method of evaluation and are expressed at a 95% confidence level.

c. Plant Design and Operational Performance

Three Mile Island, Unit 1 is a Babcox and Wilcox designed pressurized water reactor with a large volume, dry containment structure. The internal volume of the structure is approximately two million cubic feet. The concrete structure is comprised of cylindrical walls, a flat foundation mat and a shallow dome roof. The structure includes a tendon system for pre-stressing of the structure (BBRV system using 169, 0.25 inch diameter wires). The cylindrical walls are pre-stressed in the vertical and horizontal directions. The dome roof is pre-stressed using a three-way post-tensioning system. In addition to the pre-stress, mild steel reinforcing was placed in the cylinder and dome. This design is similar to the Crystal River, Unit 3 and the Zion, Unit 1 containment building designs. The containment leak rate test pressure is 50.6 PSIG.

The inside surface of the containment building is lined with a carbon steel liner to ensure leak tightness. The nominal liner thickness is 0.25 inch for the base and 0.375 inch for the remainder of the structure. Non-accessible liner seams are covered with steel test channels to permit leak testing during containment leak rate testing.

The foundation mat slab is reinforced with conventional mild steel reinforcing. The mat bears on sound rock and is nine feet thick with a concrete slab two feet thick above the bottom liner plate. The concrete mix used in the cylinder, dome and mat was designed to develop 5,000 PSI compressive strength in 28 days after pouring.

Two large openings are provided for access into the containment structure: one is a twenty-two (22) foot and four (4) inch inside diameter opening for the equipment access hatch; the other is a nine (9) foot and six (6) inch inside diameter opening for the personnel lock. Both personnel access hatches are currently leak tested.

The containment structure is protected against external corrosion by: 1) a retaining wall with a dedicated drainage system; 2) a concrete cover in excess of that required by normal construction; 3) galvanized steel construction in the construction of the conduit tendon covers, and; 4) an inboard-oriented haunch which results in only nominal tensile stresses of the outer fibers.

From an operational perspective, TMI, Unit 1 Technical Specifications require that the primary containment atmosphere pressure be maintained between 13.7 PSIA and 16.7 PSIA whenever the reactor is critical. Primary containment pressure is continuously indicated in the main control room and recorded every twelve hours as part of the Technical Specification surveillance program.

d. Containment Inspections

TMI, Unit 1 is committed to the 1992 Edition with the 1992 Addenda of the ASME Boiler and Pressure Vessel Code, Section XI.

In accordance with the ASME Code, Section XI, Exam Category E-A, Item No. E1.11, TMI, Unit 1 performs a Reactor Building containment liner general visual inspection of 100% of the accessible surfaces. This examination is required for each period during the 10-year interval. TMI, Unit 1 has completed this exam for the first period. TMI, Unit 1 is also required to perform a VT-3 (Item No. E1.12) of the Reactor Building containment liner of the accessible liner courses. This examination is required to be performed during the third period of this 10-year interval. An augmented exam of the area adjacent to the moisture barrier (i.e., between liner and concrete) is also performed. This exam is performed by creating grids in the liner area (1' X 1' grid) and ultrasonically testing this area for wall thinning. The exam is performed at this interface each period during the 10-year interval. This exam has been performed for the first period. No Section XI repairs were required.

Containment inspections also include an examination of pressure retaining bolting. Pressure retaining bolting examinations are performed in accordance with ASME Section XI, Exam Category E-G, Item No. E8.1. The TMI, Unit 1 Section XI program requires an examination of

100% of all pressure retaining bolting over the course of our 10-year interval. This includes all disassembled bolted connections and exposed surfaces of bolted connections. The exam is a VT-1. Thirty-four percent (34%) of the examinations were completed for the first period of this interval, which began on April 20, 2001. There were no unacceptable conditions identified.

NRC Information Notice 92-20 ("Inadequate Local Leak Rate Testing") addresses the inability to obtain valid local leak rate test results on penetrations which are designed with a stainless steel, two-ply bellows. There are no bellows of similar design within the TMI, Unit 1 Appendix J scope.

With regards to containment coatings, the quality assurance program for protective coatings includes the planned and systematic actions necessary to provide adequate confidence that shop or field coating work for nuclear facilities will perform satisfactorily in service and will not result in a breach of primary containment.

The quality assurance program for Protective Coatings includes the following elements:

(a) Preparation of coatings specification and procedures for generic coating materials/systems.
(b) Review and evaluation of coating manufacturers' demonstration test data and quality assurance measures for control of manufacture, identification, and performance verification of applied coating systems.
(c) Review and evaluation of supplier quality assurance measures to control storage and handling, surface preparation, application, touch-up, repair, curing and inspection of the coating systems.
(d) Training and qualification of inspection personnel in coatings inspection requirements.
(e) Supplier surveillance inspection.

Two (2) Relief Requests (RR-3 and RR-4) associated with the coatings program were approved by the USNRC for TMI, Unit 1. Approval of these Reliefs was provided in the NRC's Safety Evaluation Report (Letter from M. Gamberoni (NRC) to J. Cotton (AmerGen), dated April 27, 2000).

During T1R13 (1999), 100% of the accessible portions of the containment building liner and moisture barrier interface were examined by NDE/ISI personnel in accordance with the ASME XI IWE augmented exams. The liner showed some evidence of corrosion at the moisture barrier. In addition, the moisture barrier revealed some degradation due to corrosion. UT thickness readings were performed of the corroded areas of the liner. Furthermore, excavation of the moisture barrier was performed to assess extent of condition. Engineering determined that the extent of corrosion was limited to that area of the liner at and just above the adjoining concrete floor. TMI conservatively elected to examine the coating repairs during the next refueling outage. As such, both the liner coating and moisture barrier repairs were reexamined during T1R14 (2001) to assure repairs effectively mitigated corrosion and moisture barrier degradation.

Based on the above discussion, the ASME Section XI containment inspections and the containment coatings program provide a high degree of assurance that any degradation of the containment structure is identified and corrected before a containment leakage path is introduced.

e. Risk Assessment

As discussed in Attachment 4, the Probabilistic Safety Risk Assessment results demonstrate a very small impact on risk associated with the one time extension of the ILRT test interval to fifteen (15) years.

The analysis contained in Attachment 4 provides an assessment of the risk associated with implementing a one-time extension of the Three Mile Island, Unit 1 containment Type A integrated leak rate test (ILRT) interval from ten (10) years to fifteen (15) years. The analysis performed examined TMI, Unit 1 specific accident sequences in which the containment remains intact or the containment is impaired. The accidents are analyzed and the results are displayed according to the eight (8) EPRI accident categories defined in EPRI TR-104285:

1. Containment intact and isolated
2. Containment isolation failures due to support system or active failures
3. Type A (ILRT) related containment isolation failures
4. Type B (LLRT) related containment isolation failures
5. Type C (LLRT) related containment isolation failures
6. Other penetration related containment isolation failures
7. Containment failure due to core damage accident phenomena
8. Containment bypass

This analysis is performed using the TMI, Unit 1 internal events Level 1 and Level 2 Probabilistic Safety Assessments. The quantitative results are summarized in Table 4-1, of Attachment 4. The key results to this risk assessment are those for the ten (10) year interval (current TMI, Unit 1 condition) and the fifteen (15) year interval (proposed change).

The following is a brief summary of some of the key aspects of the ILRT test interval extension risk analysis:

- Increasing the current ten (10) year ILRT interval to fifteen (15) years results in an insignificant increase in total population dose rate, from 11.08 person-rem/year to 11.17 person-rem/year, respectively.

- The increase in the LERF risk measure, 7.13E-8/yr, is categorized as a "very small" increase per NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis."

- Likewise, the conditional containment failure probability ($CCFP_{\%}$) increases insignificantly by 0.4 percentage points.

6.0 REGULATORY ANALYSIS

The testing requirements of 10 CFR 50, Appendix J, provide assurance that leakage from the primary containment, including systems and components that penetrate the containment, does not exceed the allowable leakage values specified in Technical Specifications. The limitation on containment leakage provides assurance that the primary containment will perform its design function following plant design basis accidents.

The allowed frequency for Type A testing, as documented in NEI 94-01, is based, in part, upon a generic evaluation documented in NUREG-1493. As discussed in NUREG-1493, reducing the Type A Integrated Leak Rate Test (ILRT) testing frequency to one (1) per twenty (20) years was found to lead to imperceptible increase in risk. Additionally, while Type B and C tests identify the vast majority (greater than 95%) of all potential leakage paths, performance-based alternatives are feasible without significant risk impacts. Since leakage contributes less than 0.1 percent of overall risk under existing guidelines, the overall effect is very small.

Three Mile Island, Unit 1 has undergone seven (7) operational Type A tests in addition to the pre-operational Type A test. The results of these tests demonstrate that the Three Mile Island, Unit 1 containment structure remains an essentially leak-tight barrier and represents minimal risk to increased leakage. Additionally, the ASME Section XI containment inspections provide a high degree of assurance that any degradation of the containment structure is identified and corrected before a containment leakage path is introduced.

As discussed in Attachment 4 ("Risk Assessment for TMI Unit 1 to Support ILRT (Type A) Interval Extension Request"), the ILRT test interval extension risk analysis has concluded that:

- Increasing the current 10-year ILRT interval to 15 years results in an insignificant increase in total population dose rate, from 11.08 person-rem/year to 11.17 person-rem/year, respectively.
- The increase in the LERF risk measure, 7.13E-8/yr, is categorized as a "very small" increase per Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis."
- Likewise, the conditional containment failure probability ($CCFP_{\%}$) increases insignificantly by 0.4 percentage points.

7.0 NO SIGNIFICANT HAZARDS CONSIDERATION

We have concluded that the proposed change to the TMI, Unit 1 Technical Specifications, which will revise Technical Specification Section 6.8.5, does not involve a Significant Hazards Consideration. In support of this determination, an evaluation of each of the three (3) standards set forth in 10 CFR 50.92 is provided below.

1. The proposed Technical Specification change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed revision to Technical Specification Section 6.8.5 ("Reactor Building Leakage Rate Testing Program") involves a one-time extension to the current interval for Type A containment testing. The current test interval of ten (10) years would be extended on a one-time basis to no longer than fifteen (15) years from the last Type A test (1993). The proposed Technical Specification change does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The reactor containment is designed to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. As such, the reactor containment itself and the testing guidelines invoked to periodically demonstrate the integrity of the reactor containment exist to ensure the plant's ability to mitigate the consequences of an accident, and do not involve the prevention or identification of any precursors of an accident. Therefore, the proposed Technical Specification change does not involve a significant increase in the probability of an accident previously evaluated.

The proposed change involves only the extension of the interval between Type A containment leakage tests. Type B and C containment leakage tests will continue to be performed at the frequency currently required by plant Technical Specifications and NEI 94-01. Industry experience has shown, as documented in NUREG-1493, that Type B and C containment leakage tests have identified a very large percentage of containment leakage paths and that the percentage of containment leakage paths that are detected only by Type A testing is very small. TMI, Unit 1 ILRT test history supports this conclusion. NUREG-1493 concluded, in part, that reducing the frequency of Type A containment leak tests to once per twenty (20) years leads to an imperceptible increase in risk. Therefore, the proposed Technical Specification change does not involve a significant increase in the consequences of an accident previously evaluated.

2. <u>The proposed Technical Specification change does not create the possibility of a new or different kind of accident from any accident previously evaluated.</u>

The proposed revision to the Technical Specifications involves a one-time extension to the current interval for Type A containment testing. The reactor containment and the testing guidelines invoked to periodically demonstrate the integrity of the reactor containment exist to ensure the plant's ability to mitigate the consequences of an accident and do not involve the prevention or identification of any precursors of an accident. The proposed Technical Specification change does not involve a physical change to the plant or the manner in which the plant is operated or controlled. Therefore, the proposed Technical Specification change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. <u>The proposed Technical Specification change does not involve a significant reduction in a margin of safety.</u>

The proposed revision to Technical Specifications involves a one-time extension to the current interval for Type A containment testing. The proposed Technical Specification change does not involve a physical change to the plant or a change in the manner in which the plant is operated or controlled. The specific guidelines and conditions of the Reactor Building Leakage Rate Testing Program, as defined in Technical Specifications, exist to ensure that the degree of reactor building containment structural integrity and leak-tightness that is considered in the plant safety analysis is maintained. The overall containment leakage rate limit specified by Technical Specifications is maintained. The proposed change involves only the extension of the interval between Type A containment leakage tests. Type B and C containment leakage tests will continue to be performed at the frequency currently required by plant Technical Specifications and NEI 94-01.

NUREG-1493 concludes that reducing the Type A Integrated Leak Rate Test (ILRT) testing frequency to one per twenty (20) years was found to lead to imperceptible increase in risk. Additionally, while Type B and C tests identify the vast majority (greater than 95%) of all potential leakage paths, performance-based alternatives are feasible without significant risk impacts. Since leakage contributes less than 0.1 percent of overall risk under existing guidelines, the overall effect is very small. The TMI, Unit 1 plant specific risk analysis supports this conclusion. Therefore, the proposed Technical Specification change does not involve a significant reduction in a margin of safety.

8.0 ENVIRONMENTAL CONSIDERATION

An Environmental Assessment is not required for the one-time Technical Specification change because the proposed change to the TMI, Unit 1 Technical Specifications conforms to the criteria for "Actions Eligible for Categorical Exclusion" as specified in 10 CFR 51.22(c)(9). The proposed change will have no impact on the environment. The proposed change does not involve a Significant Hazards Consideration as discussed in the preceding section. The proposed change does not involve a significant change in the types, or a significant increase in the amounts, of any effluents that may be released offsite. In addition, the proposed change does not involve a significant increase in individual or cumulative occupational radiation exposure.

9.0 PRECEDENT

Similar ILRT extensions have been approved for Turkey Point, Units 3 and 4 (Reference 1), Crystal River Unit 3 (Reference 2), and Peach Bottom Atomic Power Station, Unit 3 (Reference 3).

10.0 REFERENCES

1. Letter from K. N. Jabbour (USNRC) to J. A. Stall (Florida Power and Light Company), "Turkey Point Units 3 and 4 – Issuance of Amendments Regarding One-Time Extension of the Integrated Leak Rate Testing Interval (TAC NOS. 3249 and MB3250), dated January 29, 2002

2. Letter from J. M. Goshen (USNRC) to D. E. Young (Crystal River Nuclear Plant), "Crystal River Unit 3 – Issuance of Amendment Regarding Containment Leakage Rate Testing Program (TAC NO. MB1439)," dated August 30, 2001

3. Letter from J. P. Boska (USNRC) to O. D. Kingsley (Exelon Nuclear), "Peach Bottom Atomic Power Station, Unit 3 – Issuance of Amendment RE: Extension of the Containment Integrated Leak Rate Test (TAC No. MB2094)," dated October 4, 2001

ATTACHMENT 2

THREE MILE ISLAND
UNIT 1

Docket No. 50-289

License No. DPR-50

MARKED UP TECHNICAL SPECIFICATION PAGE

Attached Page

TS Page 6-11c

6.8.5 Reactor Building Leakage Rate Testing Program

The Reactor Building Leakage Rate Testing Program shall be established, implemented, and maintained as follows:

A program shall be established to implement the leakage rate testing of the Reactor Building as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995

, as modified by the following exception to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J":

a. Section 9.2.3: The first Type A test performed after the September 1993 Type A test shall be performed no later than September 2008.

The peak calculated Reactor Building internal pressure for the design basis loss of coolant accident, P_{ac}, is 50.6 psig.

The maximum allowable Reactor Building leakage rate, L_a, shall be 0.1 weight percent of containment atmosphere per 24 hours at P_{ac}.

Reactor Building leakage rate acceptance criteria is $\leq 1.0\ L_a$. During the first plant startup following each test performed in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for the Type A tests.

ATTACHMENT 3

THREE MILE ISLAND
UNIT 1

Docket No. 50-289

License No. DPR-50

FINAL TECHNICAL SPECIFICATIONS CHANGES

Attached Page

TS Page 6-11c

6.8.5 Reactor Building Leakage Rate Testing Program

The Reactor Building Leakage Rate Testing Program shall be established, implemented, and maintained as follows:

A program shall be established to implement the leakage rate testing of the Reactor Building as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exception to NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J":

a. Section 9.2.3: The first Type A test performed after the September 1993 Type A test shall be performed no later than September 2008.

The peak calculated Reactor Building internal pressure for the design basis loss of coolant accident, P_{ac}, is 50.6 psig.

The maximum allowable Reactor Building leakage rate, L_a, shall be 0.1 weight percent of containment atmosphere per 24 hours at P_{ac}.

Reactor Building leakage rate acceptance criteria $\leq 1.0\ L_a$. During the first plant startup following each test performed in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for the Type A tests.

ATTACHMENT 4

THREE MILE ISLAND
UNIT 1

Docket No. 50-289

License No. DPR-50

Risk Assessment for TMI Unit 1 to Support ILRT (Type A) Interval Extension Request

ERIN® ENGINEERING AND RESEARCH, INC.

RISK ASSESSMENT FOR TMI UNIT 1 TO SUPPORT ILRT (TYPE A) INTERVAL EXTENSION REQUEST

ERIN Report No. P0467020022-2011

Prepared by: Date: 7/25/02

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TABLE OF CONTENTS

Section 1
INTRODUCTION

1.1 PURPOSE

The purpose of this analysis is to provide an assessment of the risk associated with implementing a one-time extension of the Three Mile Island Unit 1 (TMI-1) containment Type A integrated leak rate test (ILRT) interval from ten years to fifteen years. The extension would allow for substantial cost savings as the ILRT could be deferred for additional scheduled refueling outages. The risk assessment follows the guidelines from NEI 94-01 [1], the methodology used in EPRI TR-104285 [2], the NEI Interim Guidance for Performing Risk Impact Assessments In Support of One-Time Extensions for Containment Integrated Leakage Rate Test Surveillance Intervals [3], NEI Additional Information for ILRT Extensions [21], and the NRC regulatory guidance on the use of Probabilistic Risk Assessment (PRA) findings and risk insights in support of a request for a change in a plant's licensing basis as outlined in Regulatory Guide 1.174 [4].

1.2 BACKGROUND

Revisions to 10CFR50, Appendix J (Option B) allow individual plants to extend the Integrated Leak Rate Test (ILRT) Type A surveillance testing requirements from three-in-ten years to at least once per ten years. The revised Type A frequency is based on an acceptable performance history defined as two consecutive periodic Type A tests at least 24 months apart in which the calculated performance leakage was less than normal containment leakage of 1.0La (allowable leakage).

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and was established in 1995 during development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493 [5], "Performance-Based Containment Leak Test Program," September 1995, provides the technical basis to support rulemaking to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement the NRC's rulemaking basis, NEI undertook a similar study. The results of that study are documented in Electric Power Research Institute (EPRI) Research Project Report TR-104285.

The NRC report, Performance Based Leak Test Program, NUREG-1493 [5], analyzed the effects of containment leakage on the health and safety of the public and the benefits realized from the containment leak rate testing. In that analysis, it was determined that for a representative PWR plant (i.e., Surry) that containment isolation failures contribute less than 0.1 percent to the latent risks from reactor accidents. Consequently, extending the ILRT interval should not lead to any substantial increase in risk. The current analysis is being performed to confirm these conclusions based on TMI-1 specific models and available data.

Earlier ILRT frequency extension submittals have used the EPRI TR-104285 methodology to perform the risk assessment. In November and December 2001, NEI issued enhanced guidance (hereafter referred to as the NEI Interim Guidance) that builds on the TR-104285 methodology and intended to provide for more consistent submittals. [3,21] The NEI Interim Guidance was developed for NEI by EPRI using personnel who also developed the TR-104285 methodology. This TMI-1 ILRT interval extension risk assessment employs the NEI Interim Guidance methodology.

It should be noted that, in addition to ILRT tests, containment leak-tight integrity is also verified through periodic in-service inspections conducted in accordance with the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI. More specifically, Subsection IWE provides the rules and requirements for in-service inspection of Class MC pressure-retaining components and their integral attachments, and of metallic shell and penetration liners of Class CC pressure-retaining components and their integral attachments in light-water cooled plants. Furthermore, NRC regulations 10 CFR 50.55a(b)(2)(ix)(E), require licensees to conduct visual inspections of the accessible areas of the interior of the containment 3 times every 10 years. These requirements will not be changed as a result of the extended ILRT interval. In addition, Appendix J, Type B local leak tests performed to verify the leak-tight integrity of containment penetration bellows, airlocks, seals, and gaskets are also not affected by the change to the Type A test frequency. Type C tests are also not affected by the Type A test frequency change.

1.3 CRITERIA

Based on previously approved ILRT extension requests, this analysis uses the following risk metrics to characterize the change in risk associated with the one time ILRT extension:

- Change in Large Early Release Frequency (LERF)
- Change in conditional containment failure probability
- Change in population dose (person-rem/yr)

Consistent with the NEI Interim Guidance, the acceptance guidelines in Regulatory Guide 1.174 [4] are used to assess the acceptability of this one-time extension of the Type A test interval beyond that established during the Option B rulemaking of Appendix J. NRC Regulatory Guide 1.174, "An Approach for Using PRA in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis", provides NRC recommendations for using risk information in support of applications requesting changes to the license basis of the plant.

RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in core damage frequency (CDF) less than 10^{-6} per reactor year and increases in large early release frequency (LERF) less than 10^{-7} per reactor year. Since the Type A test does not impact CDF, the relevant criterion is the change in LERF. RG 1.174 also discusses defense-in-depth and encourages the use of risk analysis techniques to show that key principles, such as the defense-in-depth philosophy, are met. Therefore, the increase in the conditional containment failure probability, which helps to ensure that the defense-in-depth philosophy is maintained, will also be calculated.

In addition, based on the precedent of other ILRT extension requests [6,18,20], the total annual risk (person-rem/yr population dose) is examined to demonstrate the relative change in risk. (No threshold has been established for this parameter change.)

Section 2

METHODOLOGY

This section provides the following methodology related items:

- Brief summary of available resource documents to support the methodology
- NEI Interim Guidance for the analysis approach to be used
- General assumptions used in the evaluation
- Plant-specific inputs

2.1 General Resources Available

This section summarizes the general resources available as input. Various industry studies on containment leakage risk assessment are briefly summarized here:

1) NUREG/CR-3539 [10]
2) NUREG/CR-4220 [11]
3) NUREG-1273 [12]
4) NUREG/CR-4330 [13]
5) EPRI TR-105189 [8]
6) NUREG-1493 [5]
7) EPRI TR-104285 [2]
8) NEI Interim Guidance [3,21]

The first study is applicable because it provides one basis for the threshold that could be used in the Level 2 PSA for the size of containment leakage that is considered significant and to be included in the model. The second study is applicable because it provides a basis of the probability for significant pre-existing containment leakage at the time of a core damage accident. The third study is applicable because it is a subsequent study to

NUREG/CR-4220 that undertook a more extensive evaluation of the same database. The fourth study provides an assessment of the impact of different containment leakage rates on plant risk. The fifth study provides an assessment of the impact on shutdown risk from ILRT test interval extension. The sixth study is the NRC's cost-benefit analysis of various alternative approaches regarding extending the test intervals and increasing the allowable leakage rates for containment integrated and local leak rate tests. The seventh study is an EPRI study of the impact of extending ILRT and LLRT test intervals on at-power public risk. Finally, the eighth study includes the NEI recommended methodology for evaluating the risk associated with obtaining a one-time extension of the ILRT interval.

NUREG/CR-3539 [10]

Oak Ridge National Laboratory (ORNL) documented a study of the impact of containment leak rates on public risk in NUREG/CR-3539. This study uses information from WASH-1400 [15] as the basis for its risk sensitivity calculations. ORNL concluded that the impact of leakage rates on LWR accident risks is relatively small.

NUREG/CR-4220 [11]

NUREG/CR-4220 is a study performed by Pacific Northwest Laboratories (PNL) for the NRC in 1985. The study reviewed over two thousand LERs, ILRT reports and other related records to calculate the unavailability of containment due to leakage. The study calculated unavailabilities for Technical Specification leakages and "large" leakages. NUREG/CR-4220 assessed the "large" containment leak probability to be in the range of 1E-3 to 1E-2, with 5E-3 identified as the point estimate based on 4 PWR events in 740 reactor years and conservatively assuming a one-year duration for each event.

NUREG-1273 [12]

A subsequent NRC study, NUREG-1273, performed a more extensive evaluation of the NUREG/CR-4220 database. This assessment noted that about one-third of the reported events were leakages that were immediately detected and corrected. In addition, this study noted that local leak rate tests can detect "essentially all potential degradations" of the containment isolation system.

NUREG/CR-4330 [13]

NUREG/CR-4330 is a study that examined the risk impacts associated with increasing the allowable containment leakage rates. The details of this report have no direct impact on the modeling approach of the ILRT test interval extension, as NUREG/CR-4330 focuses on leakage rate and the ILRT test interval extension study focuses on the frequency of testing intervals. However, the general conclusions of NUREG/CR-4330 are consistent with NUREG/CR-3539 and other similar containment leakage risk studies:

> "...the effect of containment leakage on overall accident risk is small since risk is dominated by accident sequences that result in failure or bypass of containment."

EPRI TR-105189 [8]

The EPRI study TR-105189 is useful to the ILRT test interval extension risk assessment because this EPRI study provides insight regarding the impact of containment testing on shutdown risk. This study performed a quantitative evaluation (using the EPRI ORAM software) for two reference plants (a BWR-4 and a PWR) of the impact of extending ILRT and LLRT test intervals on shutdown risk.

The result of the study concluded that a small but measurable safety benefit (shutdown CDF reduced by 1E-8/yr to 1E-7/yr) is realized from extending the test interval from 3 per 10 years to 1 per 10 years.

NUREG-1493 [5]

NUREG-1493 is the NRC's cost-benefit analysis for proposed alternatives to reduce containment leakage testing intervals and/or relax allowable leakage rates. The NRC conclusions are consistent with other similar containment leakage risk studies:

- Reduction in ILRT frequency from 3 per 10 years to 1 per 20 years results in an "imperceptible" increase in risk.
- Increasing containment leak rates several orders of magnitude over the design basis would minimally impact (0.2 – 1.0%) population risk.
- Given the insensitivity of risk to the containment leak rate and the small fraction of leak paths detected solely by Type A testing, increasing the interval between integrated leak rate tests is possible with minimal impact on public risk.

EPRI TR-104285 [2]

Extending the risk assessment impact beyond shutdown (the earlier EPRI TR-105189 study), the EPRI TR-104285 study is a quantitative evaluation of the impact of extending Integrated Leak Rate Test (ILRT) and (Local Leak Rate Test) LLRT test intervals on at-power public risk. This study combined IPE Level 2 models with NUREG-1150 Level 3 population dose models to perform the analysis. The study also used the approach of NUREG-1493 in calculating the increase in pre-existing leakage probability due to extending the ILRT and LLRT test intervals.

EPRI TR-104285 used a simplified Containment Event Tree to subdivide representative core damage sequences into eight (8) categories of containment response to a core damage accident:

1. Containment intact and isolated
2. Containment isolation failures due to support system or active failures
3. Type A (ILRT) related containment isolation failures
4. Type B (LLRT) related containment isolation failures
5. Type C (LLRT) related containment isolation failures
6. Other penetration related containment isolation failures
7. Containment failure due to core damage accident phenomena
8. Containment bypass

Consistent with the other containment leakage risk assessment studies, this study concluded:

> *"These study results show that the proposed CLRT* [containment leak rate tests] *frequency changes would have a minimal safety impact. The change in risk determined by the analyses is small in both absolute and relative terms. For example, for the PWR analyzed, the change is about 0.02 person-rem per year . . ."*

NEI Interim Guidance [3,21]

NEI "Interim Guidance for Performing Risk Impact Assessments in Support of One-Time Extensions of Containment Integrated Leakage Rate Test Surveillance Intervals" [3] has been developed to provide utilities with revised guidance regarding licensing submittals. Additional information from NEI on the "Interim Guidance" was supplied in Reference [21].

A nine step process is defined which includes changes in the following areas of the previous EPRI guidance:

- Impact of extending surveillance intervals on dose
- Method used to calculate the frequencies of leakages detectable only by ILRTs
- Provisions for using NUREG-1150 dose calculations to support the population dose determination.

This NEI Guidance is used in the TMI-1 ILRT risk assessment.

2.2 NEI INTERIM GUIDANCE

This analysis uses the approach outlined in the NEI Interim Guidance. [3,21] The nine steps of the methodology are:

1. Quantify the baseline (nominal three year ILRT interval) frequency per reactor year for the EPRI accident categories of interest. Note that EPRI categories 4, 5, and 6 are not affected by changes in ILRT test frequency.
2. Determine the containment leakage rates for EPRI categories 1 and 3 where category 3 is subdivided into categories 3a and 3b for "small" and "large" isolation failures, respectively.
3. Develop the baseline population dose (person-rem) for the applicable EPRI categories.
4. Determine the population dose rate (person-rem/year) by multiplying the dose calculated in Step (3) by the associated frequency calculated in Step (1).
5. Determine the change in probability of leakage detectable only by ILRT, and associated frequency for the new surveillance intervals of interest. Note that with increases in the ILRT surveillance interval, the size of the postulated leak path and the associated leakage rate are assumed not to change, however the probability of leakage detectable only by ILRT does increase.

6. Determine the population dose rate for the new surveillance intervals of interest.

7. Evaluate the risk impact (in terms of population dose rate and percentile change in population dose rate) for the interval extension cases.

8. Evaluate the risk impact in terms of LERF.

9. Evaluate the change in conditional containment failure probability.

The first seven steps of the methodology calculate the change in dose. The change in dose is the principal basis upon which the Type A ILRT interval extension was previously granted and is a reasonable basis for evaluating additional extensions. The eighth step in the interim methodology calculates the change in LERF and compares it to the guidelines in Regulatory Guide 1.174. Because there is no change in CDF, the change in LERF forms the quantitative basis for a risk informed decision per current NRC practice, namely Regulatory Guide 1.174. The ninth and final step of the interim methodology calculates the change in containment failure probability. The NRC has previously accepted similar calculations (Ref. [7], referred to as conditional containment failure probability, CCFP) as the basis for showing that the proposed change is consistent with the defense in depth philosophy. As such this last step suffices as the remaining basis for a risk informed decision per Regulatory Guide 1.174.

2.3 ASSUMPTIONS

The following ground rules are used in the analysis:

- The TMI-1 internal events Level 1 and Level 2 PSAs are reflective of the current as-built plant and provide reasonable representative risk spectrum results for use in this analysis. External event risk results from the TMI-1 IPEEE are investigated as a sensitivity discussion.

- The base Level 3 results are obtained from a separate analysis for TMI-1, based on a generic B&W Owners Group Level 3 PSA methodology.[22]

- The use of year 2000 population data is adequate for this analysis. Scaling the year 2000 population data to July 2002 (the date of this report) would not significantly impact the quantitative results, nor would it change the conclusions.
- An evaluation of the risk impact of the ILRT on shutdown risk is addressed using the generic results from EPRI TR-105189 [8].
- Radionuclide release categories are defined consistent with the EPRI TR-104285 methodology. [2]
- Per the NEI Interim Guidance, the representative containment leakage for EPRI Category 1 sequences is 1 L_a (L_a is the Technical Specification maximum allowable containment leakage rate).
- Per the NEI Interim Guidance, the representative containment leakage for EPRI Category 3a sequences is 10 L_a. [3]
- Per the NEI Interim Guidance, the representative containment leakage for EPRI Category 3b sequences is 35 L_a. [3]
- The reduction in ILRT frequency does not impact the reliability of containment isolation valves to close in response to a containment isolation signal.

2.4 PLANT-SPECIFIC INPUTS

The TMI-1 specific information used to perform this ILRT interval extension risk assessment includes the following:

- TMI-1 Internal Events Level 1 PSA
- TMI-1 Internal Events Level 2 PSA
- TMI-1 Internal Events Level 3 PSA
- TMI-1 IPEEE
- Past TMI-1 ILRT results to demonstrate adequacy of the administrative and hardware issues.

2.4.1 TMI-1 Internal Events Level 1 PSA

The TMI-1 Level 1 PSA used as input to this analysis is characteristic of the as-built, as-operated plant. The current Level 1 PSA model is developed in Riskman. The total internal events core damage frequency (CDF) used in this analysis is 3.97E-5/yr. Table 2-1 summarizes the TMI-1 Level 1 PSA core damage frequency results by plant damage state.

2.4.2 TMI-1 Internal Events Level 2 PSA

Table 2-2 summarizes the pertinent TMI-1 internal events Level 2 PSA results in terms of release category as a function of plant damage state. As discussed in the notes to Table 2-2, release categories RC901 through RC904 refer to severe accidents with no containment failure. The total frequency of accidents in which the containment remains intact (i.e., containment leakage within Technical Specifications) is 1.87E-5/yr.

2.4.3 TMI-1 Internal Events Level 3 PSA

The NEI Interim Guidance recommends two options for calculating population dose for the EPRI categories:

- Use of NUREG-1150 dose calculations
- Use of plant-specific dose calculations

Although TMI-1 does not maintain a Level 3 PSA, a generic Level 3 PSA applicable to Babcock and Wilcox (B&W) PWR plants was performed by the B&W Owners Group. [19] The generic Level 3 PSA provided by the B&WOG in Reference [19] was enhanced in support of this ILRT risk assessment to incorporate the following:

Table 2-1

TMI-1 CORE DAMAGE FREQUENCY BY PLANT DAMAGE STATE

Plant Damage State	Core Damage Frequency (/yr)	% of CDF
PDS1A	6 61E-07	1.7%
PDS1C	9.26E-09	0 0%
PDS2B	7 49E-08	0.2%
PDS3A	8.51E-12	0.0%
PDS4A	7.86E-07	2 0%
PDS4B	3 24E-07	0.8%
PDS4C	8 85E-07	2.2%
PDS4F	1.16E-06	2.9%
PDS4L	1.34E-08	0 0%
PDS5A	9 18E-07	2.3%
PDS5B	2.39E-07	0.6%
PDS5C	7.82E-07	2.0%
PDS5F	2.04E-07	0.5%
PDS6A	6 41E-10	0.0%
PDS7A	1 05E-05	26.5%
PDS7C	4.95E-06	12.5%
PDS7D	1.16E-07	0.3%
PDS7E	9.42E-08	0.2%
PDS7F	3 46E-06	8.7%
PDS7L	4.07E-08	0.1%
PDS8A	7.28E-06	18.4%
PDS8B	1 04E-06	2.6%
PDS8C	3.57E-06	9.0%
PDS8D	9.16E-08	0.2%
PDS8E	8.10E-08	0.2%
PDS10A	4.67E-07	1.2%
PDS10C	3 59E-07	0.9%
PDS12C	1.54E-07	0 4%
PDS15A	1 10E-07	0.3%
PDS18B	6.51E-07	1.6%
PDS18C	4.10E-07	1.0%
PDS18E	8 79E-09	0.0%
PDS18F	3.78E-08	0.1%
PDS19O	1.83E-07	0.5%
Total	3.97E-05	100.0%

Notes to Table 2-1:

The TMI-1 Plant Damage States (PDSs) are defined using a two-term nomenclature. The first term is the Core Melt Bin (a numeric designator) and the second term is the Containment Safeguards and Isolation State (an alphabetical designator). These designators are summarized below:

First Term (Core Melt Bin):

1	Large LOCA, injection failure
2	Large LOCA, early recirculation failure
3	Large LOCA, late recirculation failure
4	Medium LOCA, injection failure
5	Medium LOCA, early recirculation failure
6	Medium LOCA, late recirculation failure
7	Small LOCA, injection failure, steam generators available
8	Small LOCA, recirculation failure, steam generators available
9	Small LOCA, injection failure, steam generators unavailable
10	Small LOCA, early recirculation failure, steam generators unavailable
11	Small LOCA, late recirculation failure, steam generators unavailable
12	Cycling relief valve, injection failure
13	Cycling relief valve, early recirculation failure
14	Cycling relief valve, late recirculation failure
15	Steam generator tube rupture, injection failure, steam generators unavailable
16	Steam generator tube rupture, early recirculation failure, steam generators unavailable
17	Steam generator tube rupture, late recirculation failure, steam generators unavailable
18	Steam generator tube rupture, steam generators available
19	Interfacing-systems LOCA

Second Term (Containment Safeguards and Isolation State):

A	All safeguards available, containment isolated
B	Fans available, sprays available in injection mode; sprays unavailable in recirculation mode, containment isolated
C	Fans available; sprays unavailable in injection and recirculation modes, containment isolated
D	Sprays available in injection and recirculation modes; fans unavailable, containment isolated
E	Sprays in injection mode available; fans unavailable, sprays unavailable in recirculation mode, containment isolated
F	No safeguards available, containment isolated
G	All safeguards available, small isolation failure
H	Fans available, sprays available in injection mode; sprays unavailable in recirculation mode, small isolation failure
I	Fans available; sprays unavailable in injection and recirculation modes, small isolation failure
J	Sprays available in injection and recirculation modes; fans unavailable, small isolation failure
K	Sprays in injection mode available; fans unavailable, sprays unavailable in recirculation mode, small isolation failure
L	No safeguards available, small isolation failure
M	All safeguards available, large isolation failure
N	Fans available, sprays available in injection mode; sprays unavailable in recirculation mode, large isolation failure

O	Fans available; sprays unavailable in injection and recirculation modes, large isolation failure
P	Sprays available in injection and recirculation modes; fans unavailable, large isolation failure
Q	Sprays in injection mode available; fans unavailable, sprays unavailable in recirculation mode, large isolation failure
R	No safeguards available, large isolation failure

Table 2-2
TMI-1 RADIONUCLIDE RELEASE CATEGORY FREQUENCY BY PLANT DAMAGE STATE (page 1 of 2)

	RC101	RC102	RC103	RC104	RC201	RC202	RC203	RC204	RC301	RC302	RC303	RC304	RC305	RC306	RC401	RC402	RC403	RC404	RC405	RC406	RC407
PDS1A	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PDS1C	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PDS2B	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PDS3A	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PDS4A	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PDS4B	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PDS4C	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PDS4F	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PDS4L	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PDS5A	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	1.19E-8	1.33E-9	0.0	0.0	0.0
PDS5B	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PDS5C	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PDS5F	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PDS6A	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PDS7A	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PDS7C	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PDS7D	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PDS7E	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PDS7F	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PDS7L	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PDS8A	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	3.62E-8	4.02E-9	0.0	0.0	0.0
PDS8B	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PDS8C	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PDS8D	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PDS8E	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PDS10A	0.0	3.88E-7	0.0	2.04E-8	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PDS10C	0.0	0.0	0.0	3.13E-7	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PDS12C	0.0	0.0	0.0	1.21E-7	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PDS15A	0.0	1.05E-7	0.0	5.50E-9	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PDS18B	0.0	0.0	3.26E-7	3.26E-7	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PDS18C	0.0	0.0	2.05E-7	2.05E-7	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PDS18E	0.0	0.0	4.40E-9	4.40E-9	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PDS18F	0.0	0.0	1.89E-8	1.89E-8	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
PDS19O	0.0	0.0	0.0	0.0	0.0	0.0	1.65E-7	1.83E-8	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Total	0.0	4.92E-7	5.54E-7	1.01E-6	0.0	0.0	1.65E-7	1.83E-8	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	4.81E-8	5.35E-9	0.0	0.0	0.0

Table 2-2

TMI-1 RADIONUCLIDE RELEASE CATEGORY FREQUENCY BY PLANT DAMAGE STATE (page 2 of 2)

	RC408	RC501	RC502	RC601	RC602	RC603	RC604	RC605	RC606	RC607	RC608	RC701	RC702	RC703	RC704	RC801	RC901	RC902	RC903	RC904	Total
PDS1A	0 0	3 85E-8	2 03E 9	2 62E-10	2 29E-13	1 36E-11	0 0	1 38E-11	0 0	7 81E 13	0 0	2 49E 9	2 41E-12	1 31E-10	1 14E-13	3 09E-8	5 86E 7	5 86E-10	0 0	0 0	8 61E 7
PDS1C	0 0	0 0	1 49E 10	0 0	0 0	0 0	0 0	3 45E-12	8 06E-10	1 79E 13	4 20E-11	0 0	0 0	3 27E 11	7 63E-9	5 70E-10	0 0	0 0	2 89E 11	2 70E-13	9 20E 9
PDS2B	0 0	0 0	2 71E 9	0 0	0 0	0 0	0 0	0 0	6 12E-9	0 0	3 18E-10	0 0	0 0	0 0	5 79E-8	7 86E 8	0 0	0 0	0 0	0 0	7 40E 8
PDS3A	0 0	4 96E-13	2 61E 14	3 38E-15	2 95E-18	1 75E 18	0 0	1 77E-16	0 0	9 02E 18	0 0	3 20E 14	3 11E-17	1 84E 15	1 47E-16	3 87E 13	7 55E-12	7 55E-15	0 0	0 0	8 51E 12
PDS4A	0 0	4 58E-9	2 41E 9	3 12E-10	2 45E-13	1 60E-11	0 0	1 63E-11	0 0	8 25E 13	0 0	2 95E 9	2 84E 12	1 56E 10	1 23E-13	3 67E 8	6 97E 7	6 97E-10	0 0	0 0	7 86E 7
PDS4B	0 0	0 0	1 99E 8	0 0	0 0	0 0	0 0	0 0	2 58E-8	0 0	1 34E-9	0 0	0 0	0 0	2 44E 7	3 30E-8	0 0	0 0	0 0	0 0	3 24E 7
PDS4C	0 0	0 0	1 43E 8	0 0	0 0	0 0	0 0	3 30E-10	7 70E-8	1 96E 11	4 01E-9	0 0	0 0	3 12E-9	7 29E-7	5 45E-8	0 0	0 0	2 58E 9	2 57E-11	8 85E 7
PDS4F	0 0	0 0	1 27E 8	0 0	0 0	0 0	0 0	2 40E 8	2 42E-10	1 25E 9	1 29E-11	0 0	0 0	2 27E-7	2.29E-9	4 44E-7	0 0	0 0	4 41E 7	4 44E 9	1 18E 8
PDS4L	0 0	0 0	1 47E 10	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	1 34E 8
PDS5A	0 0	5 35E-8	2 82E 9	3 64E-10	3 18E 13	1 88E-11	0 0	1 91E-11	0 0	9 73E 13	0 0	3 45E 9	3 35E-12	1 82E 10	1 59E-13	4 29E-8	8 14E 7	8 14E-10	0 0	0 0	9 18E 7
PDS5B	0 0	0 0	8 85E 9	0 0	0 0	0 0	0 0	0 0	1 95E-8	0 0	1 02E 8	0 0	0 0	0 0	1 85E 7	2 51E-8	0 0	0 0	0 0	0 0	2 39E 7
PDS5C	0 0	0 0	1 06E 8	0 0	0 0	0 0	0 0	1 48E 9	6 20E-8	7 60E 11	3 23E-9	0 0	0 0	1 38E-9	6 88E-7	7 28E-8	0 0	0 0	3 05E 8	3 08E-10	7 82E 7
PDS5F	0 0	0 0	2 24E 9	0 0	0 0	0 0	0 0	1 90E 8	1 92E-10	9 88E 10	9 94E-12	0 0	0 0	1 80E-7	1 81E-9	0 0	0 0	0 0	0 0	0 0	2 04E 7
PDS6A	0 0	3 74E-11	1 97E 12	2 54E-13	2 22E-10	1 31E-14	0 0	1 33E-14	0 0	6 79E 10	0 0	2 41E-12	2 34E-15	1 27E 13	1 11E-10	2 99E-11	5 69E-10	5 69E-13	0 0	0 0	8 41E 10
PDS7A	0 0	6 13E-7	3.23E 8	4 36E-9	3 78E 12	2 09E-11	0 0	2 29E-10	0 0	9 09E 13	0 0	3 95E 8	3 90E-11	2 08E-9	1 80E-12	4 91E-7	9 32E 8	9 32E-9	0 0	0 0	1 05E 5
PDS7C	0 0	0 0	7 97E 8	0 0	0 0	0 0	0 0	1 93E 8	4 10E-7	9 55E 11	2 03E-9	0 0	0 0	1 75E-7	3 71E-9	4 09E-7	0 0	0 0	1 44E 7	1 45E 9	4 95E 8
PDS7D	0 0	2 32E-9	1 22E 10	1 08E-8	1 07E-11	5 35E-11	4 12E-14	5 97E-10	5 39E-13	2 78E 12	0 0	9 74E 8	9 74E-11	5 12E-9	6 07E-12	0 0	0 0	0 0	0 0	0 0	1 16E 7
PDS7E	0 0	0 0	1 98E 9	0 0	0 0	0 0	0 0	0 0	9 18E-9	0 0	4 08E-11	0 0	0 0	0 0	7 40E-8	1 01E-8	0 0	0 0	0 0	0 0	9 42E 8
PDS7F	0 0	0 0	3 80E 8	0 0	0 0	0 0	0 0	3 10E 8	2 75E-7	1 54E 10	1 36E-9	0 0	0 0	2 81E-7	2 48E-8	3 32E-7	0 0	0 0	1 32E 8	1 32E-10	3 48E 8
PDS7L	0 0	0 0	4 48E 10	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	4 07E 8
PDS8A	0 0	4 24E-7	2.23E-8	3 02E-9	2 82E-12	1 45E-11	0 0	1 59E-10	0 0	9 85E 13	0 0	2 74E 8	2 63E-11	1 44E-9	1 24E-12	3 40E-7	8 45E 8	8 45E 9	0 0	0 0	7 28E 8
PDS8B	0 0	0 0	3 76E 8	0 0	0 0	0 0	0 0	0 0	8 87E-9	0 0	4 41E 10	0 0	0 0	0 0	8 02E-7	1 09E-7	0 0	0 0	0 0	0 0	1 04E 8
PDS8C	0 0	0 0	4 82E 8	0 0	0 0	0 0	0 0	8 98E 9	2 96E-7	3 42E 11	1 47E-9	0 0	0 0	9 32E-8	2 08E-8	3 32E-7	0 0	0 0	1 39E 7	1 41E 8	3 57E 8
PDS8D	0 0	1 82E-9	9 62E-11	8 47E-9	8 41E-12	4 20E-11	3 91E-14	4 48E-10	4 24E-13	2 20E 12	0 0	7 69E 8	7 88E-11	4 03E-8	4 00E-12	0 0	0 0	0 0	0 0	0 0	9 18E 8
PDS8E	0 0	0 0	1 70E 8	0 0	0 0	0 0	0 0	0 0	7 03E-9	0 0	3 49E-11	0 0	0 0	0 0	9 36E-8	8 66E-9	0 0	0 0	0 0	0 0	8 10E 8
PDS10A	0 0	3 45E-9	1 82E 10	2 34E-11	0 0	1 19E-12	0 0	1 20E-12	0 0	4 95E 14	0 0	2 22E-10	1 70E-13	1 17E-11	0 0	2 76E-9	5 23E 8	5 23E-11	0 0	0 0	4 67E 7
PDS10C	0 0	0 0	6 14E 10	0 0	0 0	0 0	0 0	0 0	4 02E-9	0 0	2 09E-10	0 0	0 0	0 0	3 80E-8	2 71E-9	0 0	0 0	0 0	0 0	3 59E 7
PDS12C	0 0	0 0	5 29E 10	0 0	0 0	0 0	0 0	0 0	2 09E-9	0 0	1 50E-10	0 0	0 0	0 0	2 74E-8	1 95E-8	0 0	0 0	0 0	0 0	1 54E 7
PDS15A	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	1 10E 7
PDS18B	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	6 51E 7
PDS18C	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	4 10E 7
PDS18E	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	8 70E 8
PDS18F	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	3 78E 8
PDS190	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	0 0	1 83E 7
Total	0 0	1 18E-6	3 42E-7	2 76E-9	2 83E 11	1 81E-10	7 73E-14	1 03E 7	1 28E-6	2 82E 8	1 57E-8	2 50E 7	2 47E-10	8 55E-7	1 17E-6	2 70E 6	1 70E 6	1 70E 8	7 70E 7	7 77E 8	3 97E 6

Notes to Table 2-2:

Forty-one (41) release categories are used in the TMI Level 2 PSA. A numbering scheme is used to separate major categories·

1: Containment Bypass with Auxiliary Building Bypass
2: Interfacing-Systems LOCA
3: Large Isolation Failures
4: Small Isolation Failures
5: Early Containment Failure
6: Late Containment Failure (Catastrophic)
7: Late Containment Failure (Benign)
8: Basemat Melt-Through
9: No Containment Failure

The general characteristics of the individual release categories are described below:

1. Release Category 1.01: containment bypass, outside the auxiliary building, without ex-vessel release of fission products, with fission product scrubbing, release begins at approximately 4 hrs
2. Release Category 1.02: containment bypass, outside the auxiliary building, without ex-vessel release of fission products, without fission product scrubbing, release begins at approximately 3 hrs
3. Release Category 1.03: containment bypass, outside the auxiliary building, with ex-vessel release of fission products, with fission product scrubbing, release begins at approximately 4 hrs
4. Release Category 1.04: containment bypass, outside the auxiliary building, with ex-vessel release of fission products, without fission product scrubbing, release begins at approximately 3 hrs
5. Release Category 2.01: containment bypass, to the auxiliary building, without ex-vessel release of fission products, with fission product scrubbing, release begins at approximately 4 hrs
6. Release Category 2.02: containment bypass, to the auxiliary building, without ex-vessel release of fission products, without fission product scrubbing, release begins at approximately 3 hrs
7. Release Category 2.03: containment bypass, to the auxiliary building, with ex-vessel release of fission products, with fission product scrubbing, release begins at approximately 4 hrs
8. Release Category 2.04: containment bypass, to the auxiliary building, with ex-vessel release of fission products, without fission product scrubbing, release begins at approximately 3 hrs
9. Release Category 3.01: large isolation failure, to the auxiliary building, without ex-vessel release of fission products, with fission product scrubbing, release begins at approximately 1.5 hrs
10. Release Category 3.02: large isolation failure, to the auxiliary building, without ex-vessel release of fission products, without fission product scrubbing, release begins at approximately 1.5 hrs
11. Release Category 3.03: large isolation failure, to the auxiliary building, with ex-vessel release of fission products, with fission product scrubbing, release begins at approximately 1.5 hrs
12. Release Category 3.04: large isolation failure, to the auxiliary building, with ex-vessel release of fission products, without fission product scrubbing, release begins at approximately 1.5 hrs

13. Release Category 3.05: large isolation failure, outside the auxiliary building, without ex-vessel release of fission products, release begins at approximately 1.5 hrs
14. Release Category 3.06: large isolation failure, outside the auxiliary building, with ex-vessel release of fission products, release begins at approximately 1.5 hrs
15. Release Category 4.01: small isolation failure, to the auxiliary building, without ex-vessel releaseof fission products, with fission product scrubbing, release begins at approximately 2.5 hrs
16. Release Category 4.02: small isolation failure, to the auxiliary building, without ex-vessel release of fission products, without fission product scrubbing, release begins at approximately 2.5 hrs
17. Release Category 4.03: small isolation failure, to the auxiliary building, with ex-vessel release of fission products, with fission product scrubbing, release begins at approximately 2.5 hrs
18. Release Category 4.04: small isolation failure, to the auxiliary building, with ex-vessel release of fission products, without fission product scrubbing, release begins at approximately 2.5 hrs
19. Release Category 4.05: small isolation failure, to the environment, without ex-vessel release of fission products, with fission product scrubbing, release begins at approximately 2.5 hrs
20. Release Category 4.06: small isolation failure, to the environment, without ex-vessel release of fission products, without fission product scrubbing, release begins at approximately 2.5 hrs
21. Release Category 4.07: small isolation failure, to the environment, with ex-vessel release of fission products, without fission product scrubbing, release begins at approximately 2.5 hrs
22. Release Category 4.08: small isolation failure, to the environment, with ex-vessel release of fission products, without fission product scrubbing, release begins at approximately 2.5 hrs
23. Release Category 5.01: early containment failure, without ex-vessel fission product release, release begins at approximately 3.25 hrs
24. Release Category 5.02: early containment failure, with ex-vessel fission product release, release begins at approximately 5.5 hrs
25. Release Category 6.01: late overpressurization, with catastrophic containment failure, without ex-vessel fission product release, without revaporization, with fission product scrubbing, release begins at approximately 45 hrs
26. Release Category 6.02: late overpressurization, with catastrophic containment failure, without ex-vessel fission product release, without revaporization, without fission product scrubbing, release begins at approximately 45 hrs
27. Release Category 6.03: late overpressurization, with catastrophic containment failure, without ex-vessel fission product release, with revaporization, with fission product scrubbing, release begins at approximately 45 hrs
28. Release Category 6.04: late overpressurization, with catastrophic containment failure, without ex-vessel fission product release, with revaporization, without fission product scrubbing, release begins at approximately 45 hrs

29. Release Category 6.05: late overpressurization, with catastrophic containment failure, with ex-vessel release of fission products, without revaporization, with fission product scrubbing, release begins at approximately 45 hrs

30. Release Category 6.06: late overpressurization, with catastrophic containment failure, with ex-vessel release of fission products, without revaporization, without fission product scrubbing, release begins at approximately 45 hrs

31. Release Category 6.07: late overpressurization, with catastrophic containment failure, with ex-vessel release of fission products, with revaporization, with fission product scrubbing, release begins at approximately 45 hrs

32. Release Category 6.08: late overpressurization, with catastrophic containment failure, with ex-vessel release of fission products, with revaporization, without fission product scrubbing, release begins at approximately 45 hrs

33. Release Category 7.01: late overpressurization, with benign containment failure, without ex-vessel fission product release, with fission product scrubbing, release begins at approximately 14.5 hrs

34. Release Category 7.02: late overpressurization, with benign containment failure, without ex-vessel fission product release, without fission product scrubbing, release begins at approximately 14.5 hrs

35. Release Category 7.03: late overpressurization, with benign containment failure, with ex-vessel release of fission products, with fission product scrubbing, release begins at approximately 14.5 hrs

36. Release Category 7.04: late overpressurization, with benign containment failure, with ex-vessel release of fission products, without fission product scrubbing, release begins at approximately 14.5 hrs

37. Release Category 8.01: containment failure from basemat melt-through, with ex-vessel release of fission products, release begins at approximately 36 hrs

38. Release Category 9.01: no containment failure, without ex-vessel fission product release, with fission product scrubbing, release begins at approximately 0.5 hrs

39. Release Category 9.02: no containment failure, without ex-vessel fission product release, without fission product scrubbing, release begins at approximately 2.5 hrs

40. Release Category 9.03: no containment failure, with ex-vessel fission product release, with fission product scrubbing, release begins at approximately 2.5 hrs

41. Release Category 9.04: no containment failure, with ex-vessel fission product release, without fission product scrubbing, release begins at approximately 2.5 hrs

- TMI-1 specific meteorological data
- TMI-1 specific population data (year 2000)
- Core radioisotope inventory parameters representative of the TMI-1 24-month fuel cycle.
- TMI-1 specific fission product release source term information for many of the release categories
- Revised baseline and sensitivity MACCS2 [17] consequence calculations

This supporting calculation is contained in TMI Calculation No. C-1101-900-E-220-178 [22] The results from that analysis are used as direct input in this risk assessment to assign 50 mile radius population doses (refer to Section 3.3). Consequently, the assumptions utilized in C-1101-900-E-220-178 are implicitly included with this analysis. The TMI-1 specific doses by release category are summarized in Table 2-3 (this table also includes the release category frequency and dose rate).

2.4.4 TMI-1 IPEEE

As discussed in Section 2.3, the TMI-1 internal events Level 1 and Level 2 PSAs are used as the primary basis for this risk assessment. However, external event risk results from the TMI-1 Individual Plant Examination of External Events (IPEEE) are used in this analysis in a sensitivity discussion to confirm that the conclusion of this analysis does not change if external events are included in the decision making process.

Refer to Appendix A of this report for further details of the TMI-1 IPEEE and the use of the IPEEE results in this risk assessment.

Table 2-3

TMI-1 50-MILE RADIUS POPULATION DOSE AND DOSE RATE AS A FUNCTION OF TMI-1 PSA RELEASE CATEGORY [22]

Release Category	Release Category Frequency (1/yr)	50-Mile Radius Population Dose (Person-Rem)	50-Mile Radius Population Dose Rate (Person-Rem/yr)
RC101	0.00E+00	5.05E+05	0.00E+00
RC102	4.92E-07	1.01E+06	4.97E-01
RC103	5.54E-07	5.06E+05	2.80E-01
RC104	1.01E-06	1.01E+06	1.02E+00
RC201	0.00E+00	3.53E+06	0.00E+00
RC202	0.00E+00	1.18E+07	0.00E+00
RC203	1.65E-07	3.71E+06	6.11E-01
RC204	1.83E-08	1.34E+07	2.45E-01
RC301	0.00E+00	1.56E+06	0.00E+00
RC302	0.00E+00	3.21E+06	0.00E+00
RC303	0.00E+00	1.89E+06	0.00E+00
RC304	0.00E+00	4.05E+06	0.00E+00
RC305	0.00E+00	3.21E+06	0.00E+00
RC306	0.00E+00	4.03E+06	0.00E+00
RC401	0.00E+00	4.23E+05	0.00E+00
RC402	0.00E+00	1.18E+06	0.00E+00
RC403	4.81E-08	6.10E+05	2.94E-02
RC404	5.35E-09	1.57E+06	8.40E-03
RC405	0.00E+00	5.42E+05	0.00E+00
RC406	0.00E+00	1.41E+06	0.00E+00
RC407	0.00E+00	9.47E+05	0.00E+00
RC408	0.00E+00	2.13E+06	0.00E+00
RC501	1.18E-06	1.15E+06	1.36E+00
RC502	3.42E-07	1.21E+06	4.14E-01
RC601	2.76E-08	3.30E+05	9.10E-03
RC602	2.63E-11	8.23E+05	2.16E-05
RC603	1.81E-10	1.15E+06	2.08E-04
RC604	7.73E-14	1.68E+06	1.30E-07

Table 2-3

TMI-1 50-MILE RADIUS POPULATION DOSE AND DOSE RATE AS A FUNCTION OF TMI-1 PSA RELEASE CATEGORY [22]

Release Category	Release Category Frequency (1/yr)	50-Mile Radius Population Dose (Person-Rem)	50-Mile Radius Population Dose Rate (Person-Rem/yr)
RC605	1.03E-07	3.37E+05	3.49E-02
RC606	1.28E-06	8.49E+05	1.09E+00
RC607	2.62E-09	1.16E+06	3.04E-03
RC608	1.57E-08	1.69E+06	2.66E-02
RC701	2.50E-07	1.27E+05	3.17E-02
RC702	2.47E-10	4.21E+05	1.04E-04
RC703	9.55E-07	1.28E+05	1.22E-01
RC704	1.17E-05	4.23E+05	4.95E+00
RC801	2.79E-06	5.78E+04	1.61E-01
RC901	1.79E-05	3.53E+02	6.33E-03
RC902	1.79E-08	8.67E+03	1.55E-04
RC903	7.70E-07	2.00E+03	1.54E-03
RC904	7.77E-09	8.76E+03	6.80E-05

2.4.5 TMI-1 Past ILRT Results

The surveillance frequency for Type A testing in NEI 94-01 under option B criteria is at least once per ten years based on an acceptable performance history (i.e., two consecutive periodic Type A tests at least 24 months apart where the calculated performance leakage rate was less than 1.0 L_a) and consideration of the performance factors in NEI 94-01, Section 11.3.

In June 1996 revised TMI-1 Technical Specifications implementing the performance-based leakage rate testing program were submitted to the NRC for approval. In May 1997, the NRC issued Amendment 201 to the TMI-1 Technical Specifications. Based on completion of two successful ILRTs at TMI-1, the current ILRT interval is once per ten years. The next Type A test for TMI-1 is currently due to be completed by October 2003 [16].

Note that the probability of a pre-existing leakage due to extending the ILRT interval is based on the industry wide historical results as discussed in the NEI Guidance document, and the only portion of TMI specific information utilized is the fact that the current ILRT interval is once per ten years.

Section 3
ANALYSIS

3.1 BASELINE ACCIDENT CATEGORY FREQUENCIES (STEP 1)

The first step of the NEI Interim Guidance is to quantify the baseline frequencies for each of the EPRI TR-104285 accident categories. This portion of the analysis is performed using the TMI-1 Level 1 and Level 2 PSA results. The results for each EPRI category are described below.

Frequency of EPRI Category 1

This group consists of all core damage accident sequences in which the containment is initially isolated and remains intact throughout the accident (i.e., containment leakage at or below maximum allowable Technical Specification leakage). Per NEI Interim Guidance, the frequency per year for this category is calculated by subtracting the frequencies of EPRI Categories 3a and 3b (see below) from the sum of all severe accident sequence frequencies in which the containment is initially isolated and remains intact (i.e., accidents classified as RC9 in the TMI-1 Level 2 PSA).

As discussed previously in Section 2.4.2, the frequency of TMI-1 severe accidents in which the containment remains intact is 1.87E-5/yr. As described below, the frequencies of the 3a and 3b categories are (5.41E-7/yr + 4.27E-7/yr) and (5.41E-8/yr + 4.27E-8/yr), respectively. Therefore, the frequency of EPRI Category 1 is calculated as (1.87E-5/yr) – (9.68E-7/yr + 9.68E-8/yr) = 1.77E-5/yr.

Frequency of EPRI Category 2

This group consists of all core damage accident sequences in which the containment isolation system function fails due to failures-to-close of large containment isolation valves (either due to support system failures; or random or common cause valve failures).

The frequency of this EPRI category is determined by summing the frequencies of the RC3-series and RC4-series release categories. As discussed in the notes to Table 2-2, the RC3-series and the RC4-series release categories represent severe accidents with containment isolation failure.

From Table 2-2, the frequency of EPRI Category 2 is (12 x 0.00) + 4.81E-8/yr + 5.35E-9/yr = 5.35E-8/yr.

Frequency of EPRI Category 3a

This group consists of all core damage accident sequences in which the containment is failed due to a pre-existing "small" leak in the containment structure that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency).

The base NEI Interim Guidance describes the calculation of a screening frequency for Category 3a and 3b by multiplying the entire plant CDF by a pre-existing containment leakage probability. In supplemental guidance provided in Reference [21], NEI discusses the conservative nature of the screening frequency calculation and describes which CDF sequence contributors can be removed from the total plant CDF to obtain a less conservative frequency estimate. Consistent with the supplement NEI Interim Guidance provided in Reference [21], the frequency per year for this category is calculated as:

Frequency 3a = [3a conditional failure probability] x [CDF - (CDF with containment failure independent of containment leakage)]

The 3a conditional failure probability (2.7E-2) value is the conditional probability of having a pre-existing "small" containment leak that is detectable only by ILRTs. This value is derived in Reference [3] and is based on data collected by NEI from 91 plants. This value is also assumed reflective of ILRT testing frequencies of 3 tests in 10 years.

The pre-existing leakage probability is multiplied by the residual core damage frequency (CDF) determined as the total CDF minus the CDF for those individual sequences that involve containment failure independent of potential pre-existing containment leakage. The following core damage accidents involve containment failure or bypass regardless of the potential existence of pre-existing containment leakage:

- Containment Bypass accidents (TMI-1 PSA RC1 and RC2)
- Severe accidents with Containment Isolation System failure (TMI-1 PSA RC3 and RC4)
- Severe accidents with containment failure due to energetic phenomena (TMI-1 PSA RC5)

Therefore, the TMI-1 PSA RC6-series, RC7-series, RC8-series and RC9-series release categories are used as the CDF on which to apply the 3a conditional failure probability.

In addition to the above, consistent with the supplemental NEI guidance in Reference [21], EPRI Category 3a is refined in this risk assessment into accidents with containment sprays available (3a Scrubbed) and with sprays unavailable (3a Unscrubbed). This refinement requires separating the RC6, RC7, RC8, and RC9 accidents into scrubbed and unscrubbed. Using the TMI-1 Level 2 PSA information summarized in Table 2-2, the contribution of scrubbed and unscrubbed sequences to RC6-9 release categories is as follows:

Release Category	Frequency of Scrubbed Accident Sequences (1/yr)	Frequency of Unscrubbed Accident Sequences (1/yr)
RC6-series	1.34E-07/yr	1.30E-06/yr
RC7-series	1.20E-06/yr	1.17E-05/yr
RC8-series	0.00	2.79E-06/yr
RC9-series	1.87E-05/yr	2.57E-08/yr
TOTAL:	2.00E-05/yr	1.58E-05/yr

Therefore, the frequency of category 3a (Scrubbed) is calculated as (2.70E-02) x (2.00E-05/yr) = 5.41E-07/yr. Likewise, the frequency of category 3a (Unscrubbed) is calculated as (2.70E-02) x (1.58E-05/yr) = 4.27E-07/yr.

Frequency of EPRI Category 3b

This group consists of all core damage accident sequences in which the containment is failed due to a pre-existing "large" leak in the containment structure that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency). In the same manner as that discussed previously for category 3a, EPRI Category 3b is refined into accidents with containment sprays available (3b Scrubbed) and with sprays unavailable (3b Unscrubbed). The frequencies are calculated in the same manner as that described above for 3a, except that the 3b pre-existing leakage probability is 2.7E-3. This value is the conditional probability of having a pre-existing "large" containment leak that is detectable only by ILRTs. This value is derived in Reference [3] and is based on data collected by NEI from 91 plants. This value is also assumed reflective of ILRT testing frequencies of 3 tests in 10 years.

Therefore, the frequency of category 3b (Scrubbed) is calculated as (2.70E-03) x (2.00E-05/yr) = 5.41E-08/yr. Likewise, the frequency of category 3b (Unscrubbed) is calculated as (2.70E-03) x (1.58E-05/yr) = 4.27E-08/yr.

Frequency of EPRI Category 4

This group consists of all core damage accident sequences in which the containment isolation function is failed due to a pre-existing failure-to-seal of Type B component(s) that would not be identifiable by an ILRT. Per NEI Interim Guidance, because this category of failures is only detected by Type B tests and not by the Type A ILRT, this group is not evaluated further in this analysis.

Frequency of EPRI Category 5

This group consists of all core damage accident sequences in which the containment isolation function is failed due to a pre-existing failure-to-seal of Type C component(s) that would not be identifiable by an ILRT. Per NEI Interim Guidance, because this category of failures is only detected by Type C tests and not by the Type A ILRT, this group is not evaluated further in this analysis.

Frequency of EPRI Category 6

This group consists of all core damage accident sequences in which the containment isolation function is failed due to "other" pre-existing failure modes (e.g., pathways left open or valves that did not properly seal following test or maintenance activities) that would not be identifiable by containment leak rate tests. Per NEI Interim Guidance, because this category of failures is not impacted by leak rate tests, this group is not evaluated further in this analysis.

Frequency of EPRI Category 7

This group consists of all core damage accident progression bins in which containment failure is induced by severe accident phenomena (e.g., overpressure). Per NEI Interim Guidance, the frequency per year for this category is based on the plant Level 2 PSA results.

As discussed in Section 2.4.2, the TMI-1 Level 2 PSA categorizes severe accidents into nine release categories (RC1 through RC9). TMI-1 containment bypass scenarios (RC1 and RC2) are assigned to EPRI Category 8. TMI-1 containment isolation system failure scenarios (RC3 and RC4) are assigned to EPRI Category 2. TMI-1 severe accidents with an intact containment (RC9) are assigned to EPRI Category 1. The remaining spectrum of TMI-1 severe accidents (RC5, RC6, RC7, and RC8) apply to EPRI Category 7.

Therefore, the frequency of EPRI Category 7 is calculated as 1.52E-6/yr + 1.43E-6/yr + 1.29E-5/yr + 2.79E-6/yr = 1.86E-5/yr.

Frequency of EPRI Category 8

This group consists of all core damage accident progression bins in which the accident is initiated by a containment bypass scenario (i.e., Steam Generator Tube Rupture or Interfacing Systems LOCA, ISLOCA). The frequency of Category 8 is determined by summing the frequencies of the RC1-series and RC2-series release categories. As discussed in the notes to Table 2-2, the RC1-series and RC2-series release categories represent severe accidents initiated by a containment bypass event. Therefore, the frequency of Category 8 is calculated as (3 x 0.0) + 4.92E-7 + 5.54E-7 + 1.01E-6 + 1.65E-7 + 1.83E-8 = 2.24E-6/yr.

Summary of Frequencies of EPRI Accident Categories

In summary, per the NEI Interim Guidance, the accident sequence frequencies that can lead to radionuclide releases to the public have been derived for accident categories defined in EPRI TR-104285. The results are summarized in Table 3-1.

3.2 CONTAINMENT LEAKAGE RATES (STEP 2)

The second step of the NEI Interim Guidance is to define the containment leakage rates for EPRI Categories 3a and 3b. As discussed earlier, EPRI Categories 3a and 3b are accidents with pre-existing containment leakage pathways ("small" and "large", respectively) that would only be identifiable from an ILRT.

The NEI Interim Guidance recommends containment leakage rates of 10La and 35La for Categories 3a and 3b, respectively. These values are consistent with previous ILRT frequency extension submittal applications. La is the plant Technical Specification maximum allowable containment leak rate. The maximum allowable overall containment leak rate (at all times between required Type A testing) for TMI-1 is less than or equal to 0.1 weight percent of the containment atmosphere per 24 hours at accident pressure (P_{AC}).

The NEI Interim Guidance describes these two recommended containment leakage rates as "conservative". The NEI recommended values of 10La and 35La are used as is in this analysis to characterize the containment leakage rates for Categories 3a and 3b.

By definition, the containment leakage rate for Category 1 (i.e., accidents with containment leakage at or below maximum allowable Technical Specification leakage) is 1.0La.

Table 3-1

BASELINE RELEASE FREQUENCY AS A FUNCTION OF EPRI CATEGORY

EPRI Category	Category Description	Frequency Estimation Methodology	Frequency (1/yr)
1	~~No Containment Failure~~: Accident sequences in which the containment remains intact and is initially isolated.	Per NEI Interim Guidance: *[Sum of TMI-1 release category RC3 and RC4 frequencies] – [Frequency EPRI Categories 3a and 3b]* *[1.87E-5/yr] – [9.68E-7/yr + 9.68E-8/yr] = 1.77E-5/yr*	1.77E-05
2	~~Containment Isolation System Failure~~: Accident sequences in which the containment isolation system function fails due to failures-to-close of large containment isolation valves (either due to support system failures, or random or common cause failures). Not affected by ILRT leak testing frequency.	Per NEI Interim Guidance: *[Sum of TMI-1 release category RC9 frequencies]*	5.35E-08
3a (Scrubbed)	~~Small Pre-Existing Failures (Scrubbed)~~: Accident sequences in which the containment is failed due to a pre-existing small leak in the containment structure that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency). Releases are scrubbed.	Per NEI Interim Guidance: *[Sum of TMI-1 frequencies for RC6 thru RC9 "scrubbed" accidents] x [2.7E-2]* *[2.00E-05/yr] x [2.70E-02] = 5.41E-7/yr*	5.41E-07
3a (Unscrubbed)	~~Small Pre-Existing Failures (Unscrubbed)~~: Accident sequences in which the containment is failed due to a pre-existing small leak in the containment structure that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency). Releases are unscrubbed.	Per NEI Interim Guidance: *[Sum of TMI-1 frequencies for RC6 thru RC9 "unscrubbed" accidents] x [2.7E-2]* *[1.58E-05/yr] x [2.70E-02] = 4.27E-7/yr*	4.27E-07

Table 3-1

BASELINE RELEASE FREQUENCY AS A FUNCTION OF EPRI CATEGORY

EPRI Category	Category Description	Frequency Estimation Methodology	Frequency (1/yr)
3b (Scrubbed)	~~Large Pre-Existing Failures (Scrubbed)~~: Accident sequences in which the containment is failed due to a pre-existing large leak in the containment structure that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency). Releases are scrubbed.	Per NEI Interim Guidance: *[Sum of TMI-1 frequencies for RC6 thru RC9 "scrubbed" accidents] x [2.7E-3]* *[2.00E-05/yr] x [2.70E-03] = 5.41E-8/yr*	5.41E-08
3b (Unscrubbed)	~~Large Pre-Existing Failures (Unscrubbed)~~: Accident sequences in which the containment is failed due to a pre-existing large leak in the containment structure that would be identifiable only from an ILRT (and thus affected by ILRT testing frequency). Releases are unscrubbed.	Per NEI Interim Guidance: *[Sum of TMI-1 frequencies for RC6 thru RC9 "unscrubbed" accidents] x [2.7E-3]* *[1.58E-05/yr] x [2.70E-03] = 4.27E-8/yr*	4.27E-08
4	~~Type B Failures~~: Accident sequences in which the containment is failed due to a pre-existing failure-to-seal of Type B components that would not be identifiable from a ILRT (and thus not affected by ILRT testing frequency).	Per NEI Interim Guidance: N/A (not affected by ILRT frequency)	n/a
5	~~Type C Failures~~: Accident sequences in which the containment is failed due to a pre-existing failure-to-seal of Type C components that would not be identifiable from a ILRT (and thus not affected by ILRT testing frequency).	Per NEI Interim Guidance: N/A (not affected by ILRT frequency)	n/a
6	~~Other Containment Isolation System Failure~~: Accident sequences in which the containment isolation system function fails due to "other" pre-existing failure modes not identifiable by leak rate tests (e.g., pathways left open or valves that did not properly seal following test or maintenance activities).	Per NEI Interim Guidance: N/A (not affected by ILRT frequency)	n/a

Table 3-1

BASELINE RELEASE FREQUENCY AS A FUNCTION OF EPRI CATEGORY

EPRI Category	Category Description	Frequency Estimation Methodology	Frequency (1/yr)
7	~~Containment Failure Due to Severe Accident Progression~~: EPRI Category 7 applies to accident sequences in which the containment is failed due to the severe accident progression. Not affected by ILRT leak testing frequency.	Per NEI Interim Guidance: *[Sum of TMI-1 release category RC5, RC6, RC7, and RC8 frequencies]*	1.86E-05
8	~~Containment Bypass Accidents~~: Accident sequences in which the containment is bypassed. Such accidents are initiated by LOCAs outside containment (i.e., SGTR or Interfacing Systems LOCA). Not affected by ILRT leak testing frequency.	Per NEI Interim Guidance: *[Sum of TMI-1 release category RC1 and RC2 frequencies]*	2.24E-06
		TOTAL:	3.97E-05

3.3 BASELINE POPULATION DOSE RATE ESTIMATES (STEPS 3-4)

The third and fourth steps of the NEI Interim Guidance are to estimate the baseline population dose (person-rem) for each EPRI category and to calculate the dose rate (person-rem/year) by multiplying the category frequencies by the estimated dose.

3.3.1 Population Dose Estimates (Step 3)

As discussed in Section 2.4.3, population dose estimates used in this risk assessment are TMI-1 specific estimates taken from TMI Calculation No. C-1101-900-E-220-178.[22] The TMI-1 specific 50-mile radius doses by TMI-1 release category are summarized in Table 2-3. Use of the 50-mile radius population dose (i.e., as opposed to dose at the site boundary or at 10-mile radius, or some other radical distance) is consistent with previously approved ILRT submittals.

Using the TMI-1 dose information summarized in Table 2-3, the population dose by EPRI accident category is defined. Three general approaches to assigning representative doses to the EPRI categories may be followed:

1. Select a suitably representative accident sequence type (i.e., TMI-1 release sub-category) for each EPRI category.
2. Select the worst-core TMI-1 release sub-category for each EPRI category.
3. Use a weighted average (on a frequency contribution basis) of the constituent release categories for each EPRI category.

The third approach is used in this risk assessment after factoring in the unique results for all of the available release categories for TMI-1. The release category assignments are consistent with the approach utilized by Crystal River (that has a similar release categorization) in their ILRT extension request [20]. This allows for the most accuracy and refinement in performing the subsequent calculations. The weighted average dose is then calculated as: (sum of constituent RC dose rates) / (sum of constituent RC frequencies). Refer to Table 2-3 for details of the TMI-1 PSA release category doses, dose rates and frequencies. The resulting baseline dose estimates as a function of EPRI category are summarized in Table 3-2.

The dose for the "no containment failure" EPRI category (EPRI Category 1) is based on the weighted average of the doses associated with the TMI-1 PSA RC9-series release categories. The RC9 series release categories represent severe accidents in which the containment is isolated and remains intact (i.e., no containment failure but containment leakage at the Technical Specification allowable leakage rate).

The dose for EPRI Category 2 (containment isolation failure) is based on the weighted average of the doses associated with the TMI-1 PSA RC3-series and RC4-series release categories. The RC3-series release categories represent severe accidents with a "small" area containment isolation failure. Similarly, the RC4-series release categories represent severe accidents with a "large" containment isolation failure.

As discussed in Section 3.1, consistent with guidance in Reference [21], EPRI Category 3a is refined into accidents with containment sprays available (3a Scrubbed) and with sprays unavailable (3a Unscrubbed). The base dose rate (i.e., containment leakage within Technical Specifications) is based on the doses associated with the TMI-1 PSA RC9-series release categories. The weighted average dose of the RC901 and RC903 release categories (which represent containment intact accidents with sprays available) is used for category 3a (Scrubbed). The weighted average dose of the RC902 and RC904 release categories (which represent containment intact accidents with sprays unavailable) is used for category 3a (Unscrubbed). Per the NEI Interim Guidance, the base dose rates for 3a (Scrubbed) and 3a (Unscrubbed) are multiplied by a factor of 10 to reflect the potential for a "small" pre-existing containment leakage pathway.

Table 3-2

TMI-1 DOSE ESTIMATES AS A FUNCTION OF EPRI CATEGORY FOR POPULATION WITHIN 50-MILE RADIUS

EPRI Category	TMI-1 PSA Release Categories Used in Characterizing Dose			50-Mile Radius Dose Applied to EPRI Category (Person-Rem)(1)
	Release Category (RC)	Sum of RC Frequencies (1/yr)	Sum of 50-Mile Radius Population Dose Rates (Person-Rem/yr)	
1	RC9 (all)	1.87E-05	8.09E-03	4.32E+02
2	RC3 (all) RC4 (all)	5.35E-08	3.78E-02	7.06E+05
3a (Scrubbed)	RC901 RC903	1.87E-05	7.87E-03	4.21E+03 (Note 2)
3a (Unscrubbed)	RC902 RC904	2.57E-08	2.23E-04	8.70E+04 (Note 2)
3b (Scrubbed)	RC901 RC903	1.87E-05	7.87E-03	1.47E+04 (Note 3)
3b (Unscrubbed)	RC902 RC904	2.57E-08	2.23E-04	3.04E+05 (Note 3)
4	n/a	n/a	n/a	n/a
5	n/a	n/a	n/a	n/a
6	n/a	n/a	n/a	n/a
7	RC5 (all) RC6 (all) RC7 (all) RC8 (all)	1.86E-05	8.20E+00	4.40E+05
8	RC1 (all) RC2 (all)	2.24E-06	2.66E+00	1.18E+06

NOTES:

(1) A weighted average approach of the constituent TMI-1 PSA release categories applicable to each EPRI accident category is used to define the dose for each EPRI category. The weighted average dose is calculated as:
(sum of constituent RC dose rates) / (sum of constituent RC frequencies). Refer to Table 2-3 for details of the TMI-1 PSA release category doses, dose rates, and frequencies.

(2) Consistent with guidance in Reference [21], EPRI Category 3a is refined into accidents with containment sprays available (3a Scrubbed) and with sprays unavailable (3a Unscrubbed). The base dose rate (i.e., containment leakage within Technical Specifications) is calculated in the manner described above for Note (1); however, the base dose rate is multiplied by a factor of 10, per the NEI Interim Guidance, to reflect the potential for a "small" pre-existing containment leakage pathway.

(3) Same comment as Note (2), except that the base dose rate is multiplied by a factor of 35, per the NEI Interim Guidance, to reflect the potential for a "large" pre-existing containment leakage pathway.

The discussion above for the 3a EPRI category applies to the 3b category, as well; except that the base dose rate is multiplied by a factor of 35 to reflect the potential for a "large" pre-existing containment leakage pathway.

As EPRI Categories 4, 5, and 6 are not affected by ILRT frequency and not analyzed as part of this risk assessment (per NEI Interim Guidance), no assignment of doses is made for these categories.

The dose for EPRI Category 7 (accidents with containment failure due to severe accident progression) is based on the weighted average of the doses associated with the TMI-1 PSA RC5-series, RC6-series, RC7-series and RC8-series release categories. These release categories represent severe accidents with containment failure occurring as a result of the severe accident progression (e.g., steam explosion, overpressurization, basemat melt-through).

The dose for the containment bypass category (EPRI Category 8), is based on the weighted average of the doses associated with the TMI-1 PSA RC1-series (SGTR accidents) and RC2-series (ISLOCA accidents) release categories.

3.3.2 Baseline Population Dose Rate Estimates (Step 4)

The baseline dose rates per EPRI accident category are calculated by multiplying the dose estimates summarized in Table 3-2 by the frequencies summarized in Table 3-1. The resulting baseline population dose rates by EPRI category are summarized in Table 3-3. As the conditional containment pre-existing leakage probabilities for EPRI Categories 3a and 3b are reflective of a 3-per-10 year ILRT frequency (refer to Section 3.1), the baseline dose rates shown in Table 3-3 are indicative of a 3-per-10 year ILRT surveillance frequency.

Table 3-3

BASELINE DOSE RATE ESTIMATES BY EPRI ACCIDENT CATEGORY FOR POPULATION WITHIN 50-MILE RADIUS

EPRI Category	Category Description	Person-Rem Within 50 miles	Accident Frequency (Per Year)	Population Dose Rate (Person-Rem/Year Within 50 miles)
1	No Containment Failure	4.32E+02	1.77E-05	7.63E-03
2	Containment Isolation System Failure	7.06E+05	5.35E-08	3.78E-02
3a (Scrubbed)	Small Pre-Existing Failures (with fission product scrubbing)	4.21E+03	5.41E-07	2.28E-03
3a (Unscrubbed)	Small Pre-Existing Failures (without fission product scrubbing)	8.70E+04	4.27E-07	3.71E-02
3b (Scrubbed)	Large Pre-Existing Failures (with fission product scrubbing)	1.47E+04	5.41E-08	7.97E-04
3b (Unscrubbed)	Large Pre-Existing Failures (without fission product scrubbing)	3.04E+05	4.27E-08	1.30E-02
4	Type B Failures	n/a	n/a	NA
5	Type C Failures	n/a	n/a	NA
6	Other Containment Isolation System Failures	n/a	n/a	NA
7	Containment Failure Due to Severe Accident Progression	4.40E+05	1.86E-05	8.20E+00
8	Containment Bypass Accidents	1.18E+06	2.24E-06	2.66E+00
		TOTAL:	3.97E-05	10.95

3.4 IMPACT OF PROPOSED ILRT INTERVAL (STEPS 5-9)

Steps 5 through 9 of the NEI Interim Guidance assess the impact on plant risk due to the new ILRT surveillance interval in the following ways:

- Determine change in probability of detectable leakage (Step 5)
- Determine population dose rate for new ILRT interval (Step 6)
- Determine change in dose rate due to new ILRT interval (Step 7)
- Determine change in LERF risk measure due to new ILRT interval (Step 8)
- Determine change in CCFP due to new ILRT interval (Step 9)

3.4.1 Change in Probability of Detectable Leakage (Step 5)

Step 5 of the NEI Interim Guidance is the calculation of the change in probability of leakage detectable only by ILRT (and associated re-calculation of the frequencies of the impacted EPRI categories). Note that with increases in the ILRT surveillance interval, the size of the postulated leak path and the associated leakage rates are assumed not to change; however, the probability of pre-existing leakage detectable only by ILRT does increase.

Per the NEI Interim Guidance, the calculation of the change in the probability of a pre-existing ILRT-detectable containment leakage is based on the relationship that relaxation of the ILRT interval results in increasing the average time that a pre-existing leak would exist undetected. Using the standby failure rate statistical model, the average time that a pre-existing containment leak would exist undetected is one-half the surveillance interval. For example, if the ILRT frequency is 1-per-10 years, then the average time that a leak would be undetected is 60 months (surveillance interval of 120 months divided by 2). The impact on the leakage probability due to the ILRT interval extension is then calculated by applying a multiplier determined by the ratio of the average times of undetection for the two ILRT interval cases.

As discussed earlier in Section 3.1, the conditional probability of a pre-existing ILRT-detectable containment leakage is divided into two categories:

- "Small" pre-existing leakage (EPRI Category 3a): 2.70E-2
- "Large" pre-existing leakage (EPRI Category 3b): 2.70E-3

The base pre-existing ILRT-detectable leakage probabilities above are reflective of a 3-per-10 year ILRT frequency. The TMI-1 plant is currently operating under a 1-per-10 year ILRT testing frequency consistent with the performance-based Option B of 10 CFR Part 50, Appendix J. [16] The baseline 3-per-10 year based leakage probabilities first need to be adjusted to reflect the current 1-per-10 year TMI-1 ILRT testing frequency, as follows:

- "Small": 2.70E-2 x [(120 months/2) / (36 months/2)] = 9.00E-2
- "Large": 2.70E-3 x [(120 months/2) / (36 months/2)] = 9.00E-3

Note that a nominal 36 month interval (i.e., as opposed to 40 months, 120/3) is used in the above adjustment calculation to reflect the 3-per-10 year ILRT frequency. This is consistent with operational practicalities and the NEI Interim Guidance.

Similarly, the pre-existing ILRT-detectable leakage probabilities for the 1-per-15 year ILRT frequency currently being pursued by TMI-1 (and the subject of this risk assessment) are calculated as follows:

- "Small": 9.00E-2 x [(180 months/2) / (120 months/2)] = 1.35E-1
- "Large": 9.00E-3 x [(180 months/2) / (120 months/2)] = 1.35E-2

Given the above adjusted leakage probabilities, the impacted frequencies of the EPRI categories are summarized below (refer to Table 3-1 for details regarding frequency calculations for the individual EPRI categories):

EPRI Category	EPRI Category Frequency as a Function of ILRT Interval		
	Baseline (3-per-10 year ILRT)	Current (1-per-10 year ILRT)	Proposed (1-per-15 year ILRT)
1	1.77E-05	1.52E-05	1.34E-05
3a (Scrubbed)	5.41E-07	1.80E-06	2.70E-06
3a (Unscrubbed)	4.27E-07	1.42E-06	2.13E-06
3b (Scrubbed)	5.41E-08	1.80E-07	2.70E-07
3b (Unscrubbed)	4.27E-08	1.42E-07	2.13E-07

Note that, per the definition of the EPRI categories, only the frequencies of Categories 1, 3a, and 3b are impacted by changes in ILRT testing frequencies.

3.4.2 Population Dose Rate for New ILRT Interval (Step 6)

The dose rates per EPRI accident category as a function of ILRT interval are summarized in Table 3-4.

3.4.3 Change in Population Dose Rate Due to New ILRT Interval (Step 7)

As can be seen from the dose rate results summarized in Table 3-4, the calculated total dose rate changes slightly from the current TMI-1 1-per-10 year ILRT interval to the proposed 1-per-15 year ILRT interval. The total dose increases from 11.08 person-rem/year to 11.17 person-rem/year (an increase of <1%).

Table 3-4

DOSE RATE ESTIMATES AS A FUNCTION OF ILRT INTERVAL FOR POPULATION WITHIN 50-MILE RADIUS

EPRI Category	Category Description	Dose Rate as a Function of ILRT Interval (Person-Rem/Yr)		
		Baseline (3-per-10 year ILRT)	Current (1-per-10 year ILRT)	Proposed (1-per-15 year ILRT)
1	No Containment Failure	7.63E-03	6.56E-03	5.79E-03
2	Containment Isolation System Failure	3.78E-02	3.78E-02	3.78E-02
3a (Scrubbed)	Small Pre-Existing Failures (with fission product scrubbing)	2.28E-03	7.59E-03	1.14E-02
3a (Unscrubbed)	Small Pre-Existing Failures (without fission product scrubbing)	3.71E-02	1.24E-01	1.86E-01
3b (Scrubbed)	Large Pre-Existing Failures (with fission product scrubbing)	7.97E-04	2.65E-03	3.98E-03
3b (Unscrubbed)	Large Pre-Existing Failures (without fission product scrubbing)	1.30E-02	4.33E-02	6.49E-02
4	Type B Failures	n/a	n/a	n/a
5	Type C Failures	n/a	n/a	n/a
6	Other Containment Isolation System Failures	n/a	n/a	n/a
7	Containment Failure Due to Severe Accident Progression	8.20E+00	8.20E+00	8.20E+00
8	Containment Bypass Accidents	2.66E+00	2.66E+00	2.66E+00
	TOTAL:	10.95	11.08	11.17

Per the NEI Interim Guidance, the change in percentage contribution to total dose rate attributable to EPRI Categories 3a and 3b is also investigated here. Using the results summarized in Table 3-4, for the current TMI-1 1-per-10 year ILRT interval, the percentage contribution to total dose rate from Categories 3a and 3b is shown to be very minor:

$$[(7.59E-3 + 1.24E-1 + 2.65E-3 + 4.33E-2) / 11.08] \times 100 = 1.6\%$$

For the proposed 1-per-15 year ILRT interval, the percentage contribution to total dose rate from Categories 3a and 3b increases slightly but remains minor:

$$[(1.14E-2 + 1.86E-1 + 3.98E-3 + 6.49E-2) / 11.17] \times 100 = 2.4\%$$

3.4.4 Change in LERF Due to New ILRT Interval (Step 8)

The risk increase associated with extending the ILRT interval involves the potential that a core damage event that normally would result in only a small radioactive release from an intact containment could in fact result in a larger release due to the increase in probability of failure to detect a pre-existing leak. Per the NEI Interim Guidance, only Category 3b sequences have the potential to result in large releases if a pre-existing leak were present. Category 3b is refined in this risk assessment into a scrubbed portion and an unscrubbed portion. The doses associated with 3b (Scrubbed) are not representative of large early releases as characterized by the LERF risk measure. As such, the change in LERF (Large Early Release Frequency) is determined by the change in the frequency of Category 3b (Unscrubbed).

Category 1 accidents are not considered as potential large release pathways because the containment remains intact. Therefore, the containment leak rate is expected to be small. Similarly, Category 3a is a "small" pre-existing leak. Other accident categories such as 2, 6, 7, and 8 could result in large releases but these are not affected by the change in ILRT interval.

The impact on the LERF risk measure due to the proposed ILRT interval extension is calculated as follows:

delta LERF = [(Frequency of EPRI Category 3b (Unscrubbed) for 1-per-15 year ILRT interval)] –
[(Frequency of EPRI Category 3b (Unscrubbed) for 1-per-10 year ILRT interval)]

= 2.13E-7/yr – 1.42E-7/yr
= 7.13E-8/yr[1]

This delta LERF of 7.13E-8/yr falls into Region III, Very Small Change in Risk, of the acceptance guidelines in NRC Regulatory Guide 1.174. Therefore, increasing the ILRT interval at TMI-1 from the currently allowed 1-per-10 years to 1-per-15 years represents a very small change in risk, and is an acceptable plant change from a risk perspective.

3.4.5 Impact on Conditional Containment Failure Probability (Step 9)

Another parameter that the NRC Guidance in Reg. Guide 1.174 states can provide input into the decision-making process is the consideration of change in the conditional containment failure probability (CCFP). The change in CCFP is indicative of the effect of the ILRT on all radionuclide releases, not just LERF. The conditional containment failure probability (CCFP) can be calculated from the risk calculations performed in this analysis.

In this assessment, based on the NEI Interim Guidance, CCFP is defined such that containment failure includes all radionuclide release end states other than the intact state

[1] The 7.13E-8/yr value, as are all calculated values in this analysis, is determined using a spreadsheet calculation of summed frequencies that contain additional significant figures beyond the 2 digits shown in the two numbers subtracted above.

(EPRI Category 1) and small pre-existing leakages (EPRI Category 3a). The conditional part of the definition is conditional given a severe accident (i.e., core damage).

Consequently, the $CCFP_{\%}$ for a given ILRT interval can be calculated by following equation:

$$CCFP_{\%} = [1 - ((1 \text{ Frequency} + 3a \text{ Frequency}) / \text{Total CDF})] \times 100\%$$

For the 10-year interval:

$$CCFP_{10} = [1 - ((1.52E\text{-}5 + 1.80E\text{-}6 + 1.42E\text{-}6) / 3.97E\text{-}5)] \times 100\%$$
$$= 53.6\%$$

And for a 15-year interval:

$$CCFP_{15} = [1 - ((1.34E\text{-}5 + 2.70E\text{-}6 + 2.13E\text{-}6) / 3.97E\text{-}5)] \times 100\%$$
$$= 54.0\%$$

Therefore, the change in the conditional containment failure probability is:

$$\Delta\, CCFP_{\%} = CCFP_{15} - CCFP_{10} = 0.4 \text{ percentage points}$$

This change in $CCFP_{\%}$ of less than 1 percentage point is insignificant from a risk perspective.

Section 4
RESULTS SUMMARY

The application of the approach based on NEI Interim Guidance [3, 21], EPRI-TR-104285 [2] and previous risk assessment submittals on this subject [6, 18, 20] have led to the quantitative results summarized in this section. These results demonstrate a very small impact on risk associated with the one time extension of the ILRT test interval to 15 years.

The analysis performed examined TMI-1 specific accident sequences in which the containment remains intact or the containment is impaired. The accidents are analyzed and the results are displayed according to the eight (8) EPRI accident categories defined in Reference [2]:

1. Containment intact and isolated
2. Containment isolation failures due to support system or active failures
3. Type A (ILRT) related containment isolation failures
4. Type B (LLRT) related containment isolation failures
5. Type C (LLRT) related containment isolation failures
6. Other penetration related containment isolation failures
7. Containment failure due to core damage accident phenomena
8. Containment bypass

This analysis is performed using the TMI-1 internal events Level 1 and Level 2 PSAs. The quantitative results are summarized in Table 4-1. The key results to this risk assessment are those for the ten year interval (current TMI-1 condition) and the fifteen year interval (proposed change). The 3-per-10 year ILRT is a baseline starting point for this risk assessment given that the pre-existing containment leakage probabilities (estimated based on industry experience - - refer to Section 3.1) are reflective of the 3-per-10 year ILRT testing.

The following is a brief summary of some of the key aspects of the ILRT test interval extension risk analysis:

- Increasing the current 10 year ILRT interval to 15 years results in an insignificant increase in total population dose rate, from 11.08 person-rem/year to 11.17 person-rem/year, respectively.
- The increase in the LERF risk measure, 7.13E-8/yr, is categorized as a "very small" increase per NRC Reg. Guide 1.174.
- Likewise, the conditional containment failure probability ($CCFP_{\%}$) increases insignificantly by 0.4 percentage points.

Table 4-1

QUANTITATIVE RESULTS AS A FUNCTION OF ILRT INTERVAL

EPRI Category	Dose (Person-Rem Within 50 miles)	Quantitative Results as a Function of ILRT Interval					
		Methodology Baseline (3-per-10 year ILRT)		Current (1-per-10 year ILRT)		Proposed (1-per-15 year ILRT)	
		Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)	Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)	Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)
1	4.32E+02	1.77E-05	7.63E-03	1.52E-05	6.56E-03	1.34E-05	5.79E-03
2	7.06E+05	5.35E-08	3.78E-02	5.35E-08	3.78E-02	5.35E-08	3.78E-02
3a (Scrubbed)	4.21E+03	5.41E-07	2.28E-03	1.80E-06	7.58E-03	2.70E-06	1.14E-02
3a (Unscrubbed)	8.70E+04	4.27E-07	3.71E-02	1.42E-06	1.24E-01	2.13E-06	1.86E-01
3b (Scrubbed)	1.47E+04	5.41E-08	7.97E-04	1.80E-07	2.65E-03	2.70E-07	3.98E-03
3b (Unscrubbed)	3.04E+05	4.27E-08	1.30E-02	1.42E-07	4.33E-02	2.13E-07	6.49E-02
4	n/a	n/a	n/a	n/a	n/a	n/a	n/a
5	n/a	n/a	n/a	n/a	n/a	n/a	n/a
6	n/a	n/a	n/a	n/a	. n/a	n/a	n/a
7	4.40E+05	1.86E-05	8.20E+00	1.86E-05	8.20E+00	1.86E-05	8.20E+00
8	1.18E+06	2.24E-06 .	2.66E+00	2.24E-06	2.66E+00	2 24E-06	2.66E+00
TOTALS:		3.97E-05	10.95	3.97E-05	11.08	3.97E-05	11.17
Increase in Dose Rate (1)					1.2E-1		8.8E-2
Increase in LERF (2)				9.94E-8		7.13E-8	
Increase in $CCFP_{\%}$ (3)				0.6		0.4	

NOTES TO TABLE 4-1:

(1) The increase in dose rate (person-rem/year) is with respect to the results for the preceding ILRT interval, as presented in the table. For example, the increase in dose rate for the proposed 1-per-15 ILRT is calculated as: total dose rate for 1-per-15 year ILRT, 11.17, minus total dose rate for 1-per-10 year ILRT, 11.08.

(2) The increase in Large Early Release Frequency (LERF) is with respect to the results for the preceding ILRT interval, as presented in the table. As discussed in Section 3.4.4 of the report, the change in LERF is determined by the change in the accident frequency of EPRI Category 3b (Unscrubbed). For example, the increase in LERF for the proposed 1-per-15 ILRT is calculated as: 3b (Unscrubbed) frequency for 1per-15 year ILRT, 2.13E-07/yr, minus 3b (Unscrubbed) frequency for 1-per-10 year ILRT, 1.42E-07/yr.

(3) The increase in the conditional containment failure probability ($CCFP_{\%}$) is with respect to the results for the preceding ILRT interval, as presented in the table. As discussed in Section 3.4.5, $CCFP_{\%}$ is calculated as:

$$CCFP_{\%} = [1 - ((\text{Category 1 Frequency} + \text{Category 3a Frequency}) / \text{CDF})] \times 100\%$$

Section 5
CONCLUSIONS

5.1 QUANTITATIVE CONCLUSIONS

The conclusions from the risk assessment of the one time ILRT extension can be characterized by the risk metrics used in previously approved ILRT test interval extensions. These include:

- Change in LERF
- Change in conditional containment failure probability
- Change in population dose

Based on the results from Sections 3 and 4, the main conclusion regarding the impact on plant risk associated with extending the Type A ILRT test frequency from ten years to fifteen years is:

> Reg. Guide 1.174 [4] provides guidance for determining the risk impact of plant-specific changes to the licensing basis. Reg. Guide 1.174 defines very small changes in risk as resulting in increases of CDF below 10^{-6}/yr and increases in LERF below 10^{-7}/yr. Since the ILRT does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A ILRT test interval from once-per-ten years to once-per-fifteen years is 7.13E-8/yr. Guidance in Reg. Guide 1.174 defines very small changes in LERF as below 10^{-7}/yr. Therefore, increasing the TMI-1 ILRT interval from 10 to 15 years results in a very small change in risk, and is an acceptable plant change from a risk perspective.

The change in conditional containment failure probability (CCFP) is also calculated as an additional risk measure to demonstrate the impact on defense-in-depth. The $\Delta CCFP_{\%}$ is found to be very small and represents a negligible change in the TMI-1 defense-in-depth.

The change in population dose is also reported consistent with previously approved ILRT interval extension requests. The change in population dose rate from the current 1/10 year ILRT frequency to 1/15 year frequency is insignificant, an increase of 0.8%.

5.2 RISK TRADE-OFF

The performance of an ILRT introduces risk. An EPRI study of operating experience events associated with the performance of ILRTs has indicated that there are real risk impacts associated with the setup and performance of the ILRT during shutdown operation [8]. While these risks have not been quantified for TMI-1, it is judged that there is a positive (yet unquantified) safety benefit associated with the avoidance of frequent ILRTs.

The safety benefits relate to the avoidance of plant conditions and alignments associated with the ILRT which place the plant in a less safe condition leading to events related to drain down or loss of shutdown cooling. Therefore, while the focus of this evaluation has been on the negative aspects, or increased risk, associated with the ILRT extension, there are in fact some positive safety benefits.

5.3 EXTERNAL EVENTS IMPACT

The impact of external events on this ILRT risk assessment is summarized in this section (refer to Appendix A for further detail).

Given the characteristics of this proposed plant change (i.e., ILRT interval extension), specific quantitative information regarding the impact on external event hazard risk measures is not a significant decision making input. The proposed ILRT interval extension impacts plant risk in a very specific and limited way, that is, it impacts a subset of accident sequences in which the probability of a pre-existing containment leak is the initial containment failure mode given a core damage accident. This impact is manifested in the plant risk profile in a similar manner for internal events and external events.

Although it is not possible at this time to incorporate realistic quantitative risk assessments of all external event hazards into this assessment (i.e., the spectrum of external hazards have been evaluated in the TMI-1 IPEEE to varying levels of screening and conservatism), the quantitative results of the TMI-1 IPEEE have been evaluated as a sensitivity case to show that the conclusions of this analysis would not be altered if external events were explicitly considered.

The quantitative consideration of external hazards is discussed in more detail in Appendix A of this report. The assessment of the external events uses the results of the TMI-1 IPEEE and does not modify the IPEEE analysis and frequency results, but maintains the conservative nature of the risk results. As can be seen from the Appendix A, if the conservative results of the TMI-1 IPEEE are used directly in this assessment, the change in LERF will increase to a delta value of 2.61E-7/yr. This delta LERF falls within NRC RG 1.174 Region II ("Small Changes" in risk). As such, consistent with RG 1.174, the total TMI-1 LERF was estimated and shown to be less than the RG 1.174 limit of 1E-5/yr (refer to Appendix A).

Therefore, incorporating the conservative TMI-1 IPEEE external events accident sequence results into this analysis does not change the conclusion of this risk assessment (i.e., increasing the TMI-1 ILRT interval from 10 to 15 years is an acceptable plant change from a risk perspective).

5.4 PREVIOUS ASSESSMENTS

The NRC in NUREG-1493 [5] has previously concluded that:

- Reducing the frequency of Type A tests (ILRTs) from three per 10 years to one per 20 years was found to lead to an imperceptible increase in risk. The estimated increase in risk is very small because ILRTs identify only a few potential containment leakage paths that cannot be identified by Type B and C testing, and the leaks that have been found by Type A tests have been only marginally above existing requirements.

- Given the insensitivity of risk to containment leakage rate and the small fraction of leakage paths detected solely by Type A testing, increasing the interval between integrated leakage-rate tests is possible with minimal impact on public risk. The impact of relaxing the ILRT frequency beyond one in 20 years has not been evaluated.

The findings for TMI-1 confirm the above general findings on a plant specific basis when considering (1) TMI-1 severe accident risk profile, (2) the TMI-1 containment failure modes, and (3) the local population surrounding the TMI-1 site.

Section 6

REFERENCES

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Appendix A
EXTERNAL EVENT ASSESSMENT

A.1 INTRODUCTION

This appendix discusses the external events assessment in support of the TMI-1 ILRT frequency extension risk assessment.

External hazards were evaluated in the TMI-1 Individual Plant Examination of External Events (IPEEE) Submittal in response to the NRC IPEEE Program. The IPEEE Program was a one-time review of external hazard risk to identify potential plant vulnerabilities and to understand severe accident risks. TMI-1 does not currently maintain external event PSA models and associated documentation. Although the external event hazards in the TMI-1 IPEEE were evaluated to varying levels of conservatism, the results of the TMI-1 IPEEE are nonetheless used in this risk assessment to provide a comparative understanding of the impact of external hazards on the conclusions of this ILRT interval extension risk assessment.

The TMI-1 IPEEE study evaluated the following categories of external hazards:

- Seismic Events
- Internal Fires
- External Floods
- High Winds
- Other (e.g., aircraft impacts, nearby facility hazards, etc.)

Consistent with NRC guidance for the IPEEE Program (NUREG-1407), TMI-1 employed probabilistic screening approaches to screen out many hazards from unnecessary detailed analysis, and analyzed the more significant hazards with further probabilistic analysis.

The TMI-1 IPEEE results are summarized in Table A-1. As can be seen from Table A-1, seismic events, internal fires, and external floods contribute 99% to the plant risk resulting from external hazards. As such, these three specific hazards are reviewed as part of this ILRT risk assessment, and the other external event hazards are reasonably assumed not to impact the results or conclusions of this risk assessment.

The seismic event, internal fires and external flood analyses of the TMI-1 IPEEE are summarized below.

A.2 TMI-1 IPEEE SEISMIC ANALYSIS

Seismic event hazards at TMI-1 were evaluated in the TMI-1 IPEEE using a seismic PRA approach (i.e., as opposed to the deterministic-based seismic margins approach available as an alternative methodology for many licensees). As such, although the analysis is conservative, quantitative insights regarding the relative frequency and associated characteristics of seismic-induced severe accidents is available from the TMI-1 IPEEE.

The seismic-induced core damage frequency results from the TMI-1 IPEEE, as a function of Plant Damage State (PDS), are summarized in Table A-2. As can be seen from Table A-2, over 75% of the seismic accident sequences result in PDS7F (small LOCA, no injection, steam generators available, no containment safeguards available, containment isolated).

The TMI-1 IPEEE does not analyze explicitly release category information. As such, a breakdown of release category (RC) frequency results for seismic events is not available from the TMI-1 IPEEE. However, PDS vs RC relationship information available from the TMI-1 internal events PSA can be used to provide a reasonable representation of the spectrum of seismic-induced radionuclide releases.

Table A-1

RESULTS OF TMI-1 IPEEE
- Contributions to External Event CDF -

External Event	Contribution to External Event CDF (%)
Seismic Events	23.6
Internal Fires	15.8
External Floods	59.6
High Winds	0.6
Aircraft Crash	0.3
Hazardous Chemical Accidents	0.1
Other	negligible

Table A-2

SUMMARY OF TMI-1 IPEEE SEISMIC ANALYSIS RESULTS

PDS	PDS Definition	External Events CDF	
		PDS Frequency (1/yr)	Contribution (%)
PDS7F	Small LOCA, injection failure, steam generators available, no containment safeguards available, containment isolated	2.43E-5	75.6
PDS4F	Small LOCA, injection failure, no containment safeguards available, containment isolated	6.42E-6	20.0
PDS7R	Small LOCA, injection failure, steam generators available, no containment safeguards available, large containment isolation failure	5.47E-7	1.7
PDS4C	Small LOCA, injection failure, fans available, sprays unavailable in injection and recirculation modes, containment isolated	1.86E-7	0.58
PDS7L	Small LOCA, injection failure, steam generators available, no containment safeguards available, small containment isolation failure	1.77E-7	0.55
PDS4R	Small LOCA, injection failure, no containment safeguards available, large containment isolation failure	1.46E-7	0.45
PDS1F	Large LOCA, injection failure, no containment safeguards available, containment isolated	1.43E-7	0.44
PDS4E	Small LOCA, injection failure, sprays in injection mode available, fans unavailable, sprays unavailable in recirculation mode, containment isolated	9.47E-8	0.29
All Others	All other PDS categories.	1.25E-7	0.39
	TOTAL:	3.21E-5	100

Using the PDS vs RC ratios from the TMI-1 internal events Level 2 PSA information provided in Table 2-2 of this report, release category frequencies representative of the TMI-1 IPEEE seismic analysis are summarized in Table A-3. This information is used in Section A.5 of this appendix to provide insight into the impact of external hazard risk on the conclusions of this ILRT risk assessment.

A.3 TMI-1 IPEEE INTERNAL FIRES ANALYSIS

Internal fire hazards at TMI-1 were evaluated in the TMI-1 IPEEE using a probabilistic screening approach based on the EPRI Fire Induced Vulnerability Evaluation (FIVE) methodology. As such, although the analysis is conservative, quantitative insights regarding the relative frequency and associated characteristics of fire-induced severe accidents is available from the TMI-1 IPEEE.

The core damage frequency results from the TMI-1 IPEEE for the unscreened fire areas, as a function of Plant Damage State (PDS), are summarized in Table A-4. As can be seen from Table A-4, over 50% of the fire accident sequences result in PDS7F (small LOCA, no injection, steam generators available, no containment safeguards available, containment isolated).

The TMI-1 IPEEE does not analyze explicitly release category information. As such, a breakdown of release category (RC) frequency results for fire events is not available from the TMI-1 IPEEE. However, PDS vs RC relationship information available from the TMI-1 internal events PSA can be used to provide a reasonable representation of the spectrum of fire-induced radionuclide releases. This information is also summarized in Table A-4.

Table A-3

PDS AND ESTIMATED RELEASE CATEGORY FREQUENCIES FOR SEISMIC-INDUCED ACCIDENTS

TMI-1 IPEEE Seismic Results		Estimated Release Category Frequencies [1] (1/yr)							
PDS	PDS Frequency (1/yr)	RC306	RC502	RC605	RC606	RC703	RC704	RC801	RC903
PDS4F	6.42E-6	---	6.42E-8	1.28E-7	---	1.28E-6	---	2.50E-6	2.44E-6
PDS7F	2.43E-5	---	2.43E-7	2.43E-7	1.94E-6	1.94E-6	1.75E-5	2.43E-6	---
PDS7R [2] [3]	1.38E-6	1.38E-6	---	---	---	---	---	---	---
TOTALS:	3.21E-5	1.38E-6	3.07E-7	3.71E-7	1.94E-6	3.23E-6	1.75E-5	4.93E-6	2.44E-6

NOTES:

(1) Seismic Release Category frequencies obtained by multiplying the seismic PDS frequency from the TMI-1 IPEEE by the TMI-1 internal events PSA PDS-to-RC ratios obtained from the information in Table 2-2 of this report. Only those Release Categories with non-zero frequencies are summarized in this table.

(2) For simplicity, all the negligible seismic contributors summarized in Table A-2 are summed into one PDS (i.e., PDS7R) for this risk assessment.

(3) Release Category fequency information for the PDS7R Plant Damage State is not available from the TMI-1 PSA. Based on the definitions of PDS7R and RC306 (refer to Tables 2-1 and 2-2 of this report for these definitions), the entire seismic CDF frequency of PDS7R is reasonably assigned to Release Category RC306.

Table A-4

PDS AND ESTIMATED RELEASE CATEGORY FREQUENCIES FOR FIRE-INDUCED ACCIDENTS

TMI-1 IPEEE Internal Fire Results		Estimated Release Category Frequencies [1] (1/yr)								
PDS	PDS Frequency (1/yr)	RC501	RC502	RC605	RC606	RC703	RC704	RC801	RC901	RC903
PDS4C [2]	3.24E-6	---	6.48E-8	---	2.92E-7	---	2.66E-6	2.27E-7	---	---
PDS4F	1.73E-6	---	1.73E-8	3 46E-8	---	3.46E-7	---	6.74E-7	---	6.57E-7
PDS7C	1.30E-6	---	2.59E-8	---	1.04E-7	5.18E-8	9.72E-7	1.04E-7	---	3.89E-8
PDS7E	1.51E-6	---	3.02E-8	---	1.36E-7	---	1.18E-6	1.66E-7	---	---
PDS7F	1.12E-5	---	1.12E-7	1.12E-7	8.99E-7	8.99E-7	8.09E-6	1.12E-6	---	---
PDS8A	1.51E-6	9.07E-8	---	---	---	---	---	7.56E-8	1.35E-6	---
PDS8E	1.08E-6	---	2.16E-8	---	9.72E-8	---	8.42E-7	1.19E-7	---	---
TOTALS:	2.16E-5	9.07E-8	2.72E-7	1.47E-7	1.53E-6	1.30E-6	1.37E-5	2.49E-6	1.35E-6	6.96E-7

NOTES TO TABLE A-4:

(1) Fire Release Category frequencies obtained by multiplying the fire PDS frequency from the TMI-1 IPEEE by the TMI-1 internal events PSA PDS-to-RC ratios obtained from the information in Table 2-2 of this report. Only those Release Categories with non-zero frequencies are summarized in this table.

(2) For simplicity, all the minor fire contributors are summed into one PDS (i.e., PDS4C) for the purposes of this risk assessment. This is the most conservative PDS assignment as it translates into the largest contribution to the delta LERF calculation.

A.4 TMI-1 IPEEE EXTERNAL FLOODS ANALYSIS

External flooding hazards at TMI-1 were evaluated in the TMI-1 IPEEE using probabilistic accident sequence analysis. As such, , although the analysis is conservative, quantitative insights regarding the relative frequency and associated characteristics of external flooding severe accidents is available from the TMI-1 IPEEE.

The external flooding core damage frequency results from the TMI-1 IPEEE are summarized in Table A-5. Various individual external flooding accident sequences are evaluated in the TMI-1 IPEEE external flooding analysis, the results in Table A-5 present the summed frequency results as a function of external flood initiator. As can be seen from Table A-5, the TMI-1 IPEEE external flooding analysis evaluated three general categories of external floods:

1. External flood elevations below Elevation 305' (site will not be impacted unless dike fails)
2. External flood elevations above Elevation 310' (critical plant structures will be flooded despite implementation of flood protective measures per plant procedures)
3. External flood elevations between Elevations 305' – 310'

The TMI-1 IPEEE external flooding analysis does not explicitly assign Plant Damage State categories to the analyzed external flooding core damage sequences. However, accident sequence descriptions provided in the TMI-1 IPEEE documentation provide sufficient information in most cases to allow PDS categories to be assigned. Using the accident sequence description information in the TMI-1 IPEEE and the TMI-1 PDS definitions summarized in Table 2-1 of this report, the TMI-1 IPEEE external flooding results are summarized in Table A-6 as a function of PDS.

Table A-5

SUMMARY OF TMI-1 IPEEE EXTERNAL FLOODING ANALYSIS RESULTS
(External Flooding CDF as a Function of External Flooding Initiator)

External Flooding Initiator	Initiator Frequency (1/yr)	External Flooding CDF	
		Frequency (1/yr)	Contribution (%)
External flood elevations below Elevation 305' (site will not be impacted unless dike fails)	2.50E-3	2.50E-7	0.3
External flood elevations above Elevation 310' (critical plant structures will be flooded despite implementation of flood protective measures per plant procedures)	2.50E-4	6.37E-5	78.6
External flood elevations between Elevations 305' – 310'	4.50E-4	1.71E-5	21.1
	TOTAL:	8.10E-5	100

Table A-6

ESTIMATED RELEASE CATEGORY FREQUENCIES FOR EXTERNAL FLOODING ACCIDENTS

TMI-1 IPEEE Ext. Flooding Results		Estimated Release Category Frequencies [(1)] (1/yr)										
PDS	PDS Frequency (1/yr)	RC102	RC104	RC501	RC502	RC605	RC606	RC703	RC704	RC801	RC901	RC903
PDS5A	6.36E-6			3.82E-7						3.18E-7	5.66E-6	
PDS7A	8.65E-8			5.19E-9						4.33E-9	7.70E-8	
PDS7C	1.72E-5				3.44E-7		1.38E-6	6.88E-7	1.29E-5	1.38E-6		5.16E-7
PDS7F	3.95E-5				3.95E-7	3.95E-7	3.16E-6	3.16E-6	2.84E-5	3.95E-6		
PDS10A	1.79E-5	1.49E-5	7.16E-7	1.79E-7						1.79E-7	1.97E-6	
TOTALS:	8.10E-5	1.49E-5	7.16E-7	5.66E-7	7.39E-7	3.95E-7	4.54E-6	3.85E-6	4.13E-5	5.83E-6	7.71E-6	5.16E-7

NOTES:

(1) External Flooding Release Category frequencies obtained by multiplying the external flooding PDS frequency by the TMI-1 internal events PSA PDS-to-RC ratios obtained from the information in Table 2-2 of this report. Only those Release Categories with non-zero frequencies are summarized in this table.

(2) Based on the accident sequence description information in the TMI-1 IPEEE and the TMI-1 PDS definitions summarized in Table 2-1 of this report, the assignment of the external flooding accident sequences to PDS category is performed as follows:

- >310' El. Floods: 56% of CDF (of sequences due to this IE) assigned to PDS7F; 28% to PDS10A; and 16% to PDS7C
- <305' El. Floods: 56% of CDF (of sequences due to this IE) assigned to PDS7F; 28% to PDS10A; and 16% to PDS7C
- 305' – 310' El. Flood: Seqs. A & B assigned to PDS5A, seqs. C & D to PDS7C, seq. E to PDS7F, and seq. F to PDS7A.

Table A-6 also provides a breakdown of external flooding PDS versus release category. The TMI-1 IPEEE does not analyze explicitly release category information. As such, a breakdown of release category (RC) frequency results for external flooding events is not available from the TMI-1 IPEEE. However, PDS vs RC relationship information available from the TMI-1 internal events PSA can be used to provide a reasonable representation of the spectrum of seismic-induced radionuclide releases. Using the PDS vs RC ratios from the TMI-1 internal events Level 2 PSA information provided in Table 2-2 of this report, release category frequencies representative of the TMI-1 IPEEE external flooding analysis are summarized in Table A-6. This information is used in Section A.5 of this appendix to provide insight into the impact of external hazard risk on the conclusions of this ILRT risk assessment.

A.5 IMPACT OF EXTERNAL HAZARD RISK ON ILRT RISK ASSESSMENT

The NEI Interim Guidance methodology performed in Section 3 of this report is re-performed here including, in addition to internal event information, the TMI-1 IPEEE based external event release category information discussed in the previous sections.

A.5.1 Baseline EPRI Category Frequencies (Including TMI-1 External Events)

The baseline EPRI category frequencies are estimated here in the same manner as that described in Section 3 of this report, except that the TMI-1 IPEEE based estimates of external event initiated release category contributions are included.

Frequency of EPRI Category 1

Per NEI Interim Guidance, the frequency per year for this category is calculated by subtracting the frequencies of EPRI Categories 3a and 3b (see below) from the sum of all severe accident sequence frequencies in which the containment is initially isolated and remains intact (i.e., accidents classified as RC9 in the TMI-1 Level 2 PSA).

The frequency of TMI-1 severe accidents in which the containment remains intact is 1.87E-5/yr (due to internal events) + 1.27E-5/yr (due to external events) = 3.14E-5/yr. As described below, the frequencies of the 3a and 3b categories are 2.36E-6/yr and 2.36E-7/yr, respectively. Therefore, the frequency of EPRI Category 1 is calculated as (3.14E-5/yr) – (2.36E-6/yr + 2.36E-7/yr) = 2.88E-5/yr.

Frequency of EPRI Category 2

The frequency of this EPRI category is determined by summing the frequencies of the RC3-series and RC4-series release categories. As discussed in the notes to Table 2-2, the RC3-series and the RC4-series release categories represent severe accidents with containment isolation failure. The internal events contribution is 5.35E-8/yr (refer to Section 3.1). The external events contribution is 1.38E-6/yr. Therefore, the frequency of EPRI Category 2 is 5.35E-8/yr + 1.38E-6/yr = 1.43E-6/yr.

Frequency of EPRI Category 3a

As discussed in more detail in Section 3.1, the frequency per year for this category is calculated as:

Frequency 3a = [3a conditional failure probability] x [CDF - (CDF with containment failure independent of containment leakage)]

Also as discussed in Section 3.1, EPRI Category 3a is refined in this risk assessment into accidents with containment sprays available (3a Scrubbed) and with sprays unavailable (3a Unscrubbed).

In addition to the above, the refinement (consistent with supplemental NEI guidance) is made for external event contributors to exclude External Flooding scenarios from the frequency calculation of category 3a. Per TMI-1 Emergency Procedure 1202-32, Flood, a plant shutdown will be initiated at a river elevation of 302' El., approximately 8 feet below the TMI-1design basis external flooding elevation (310' El.). Per the TMI-1 design basis Probable Maximum Flood characteristics, plant shutdown would be initiated about 27 hours prior to flood elevation reaching 310' El.. In the TMI-1 IPEEE a worst-case hurricane was also considered, and in this case the design basis flood elevation would be exceeded in approximately 5 hours after shutdown was initiated. As such, given the slow-developing nature of external flooding scenarios, such scenarios would not result in LERF releases because of the delayed time to core damage.

Therefore, the frequency of category 3a (Scrubbed) is calculated as (2.70E-02) x (2.00E-05/yr [internal events contribution] + 9.55E-06/yr [external events contribution]) = 7.98E-07/yr. Likewise, the frequency of category 3a (Unscrubbed) is calculated as (2.70E-02) x (1.58E-05/yr [internal events contribution] + 4.21E-05/yr [external events contribution]) = 1.56E-06/yr.

Frequency of EPRI Category 3b

In the same manner as that discussed previously for category 3a, EPRI Category 3b is refined into accidents with containment sprays available (3b Scrubbed) and with sprays unavailable (3b Unscrubbed). The frequencies are calculated in the same manner as that described above for 3a, except that the 3b pre-existing leakage probability is 2.7E-3.

Therefore, the frequency of category 3b (Scrubbed) is calculated as (2.70E-03) x (2.00E-05/yr [internal events contribution] + 9.55E-06/yr [external events contribution]) = 7.98E-08/yr. Likewise, the frequency of category 3b (Unscrubbed) is calculated as (2.70E-03) x

(1.58E-05/yr [internal events contribution] + 4.21E-05/yr [external events contribution]) = 1.56E-07/yr.

Frequency of EPRI Category 4

Per NEI Interim Guidance, because this category of failures is only detected by Type B tests and not by the Type A ILRT, this group is not evaluated further in this analysis.

Frequency of EPRI Category 5

Per NEI Interim Guidance, because this category of failures is only detected by Type C tests and not by the Type A ILRT, this group is not evaluated further in this analysis.

Frequency of EPRI Category 6

Per NEI Interim Guidance, because this category of failures is not impacted by leak rate tests, this group is not evaluated further in this analysis.

Frequency of EPRI Category 7

This group consists of all core damage accident progression bins in which containment failure is induced by severe accident phenomena (e.g., overpressure). As discussed in Section 3.1, the frequency of this category is calculated by summing the frequencies of release categories RC5, RC6, RC7, and RC8. The internal events contribution is 1.86E-5/yr (refer to Section 3.1). The external events contribution is 1.05E-4/yr. Therefore, the frequency of EPRI Category 7 is 1.86E-5/yr + 1.05E-4/yr = 1.24E-4/yr.

Frequency of EPRI Category 8

As discussed in Section 3.1, the frequency of EPRI category 8 is calculated by summing the frequencies of release categories RC1 and RC2. The internal events contribution is 2.24E-6/yr (refer to Section 3.1). The external events contribution is 1.56E-5/yr. Therefore, the frequency of EPRI Category 8 is 2.24E-6/yr + 1.56E-5/yr = 1.78E-5/yr.

A.5.2 Dose Rate Estimates (Including TMI-1 External Events)

The baseline dose as a function of EPRI category are estimated in the same manner as that discussed in Section 3.3.1. As the doses are calculated on a weighted average contribution to dose rate basis, the doses estimated when external event accident frequencies are included vary slightly from that presented in the base analysis (internal events only) in Section 3. The baseline doses and dose rates (incorporating external events) as function of EPRI category are summarized in Table A-7.

A.5.3 Change in LERF (Including TMI-1 External Events)

As discussed in Section 3.4.4, the change in LERF associated with extending the ILRT interval is determined by the change in the frequency of EPRI category 3b (Unscrubbed). As can be seen from Table A-7, the increase in the LERF risk measure due to extending the ILRT from 1-per-10 years to 1-per-15 years is 2.61E-7/yr.

Comparison to RG 1.174 Acceptance Guidelines

NRC Regulatory Guide 1.174, "An Approach for Using PRA in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis", provides NRC recommendations for using risk information in support of applications requesting changes to the license basis of

Table A-7

QUANTITATIVE RESULTS AS A FUNCTION OF ILRT INTERVAL (INCLUDING EXTERNAL EVENTS)

EPRI Category	Dose (Person-Rem Within 50 miles)	Quantitative Results as a Function of ILRT Interval					
		Methodology Baseline (3-per-10 year ILRT)		Current (1-per-10 year ILRT)		Proposed (1-per-15 year ILRT)	
		Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)	Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)	Accident Frequency (per year)	Population Dose Rate (Person-Rem/Year Within 50 miles)
1	5.92E+02	2.88E-05	1.71E-02	2.28E-05	1.35E-02	1.84E-05	1.09E-02
2	3.91E+06	1.43E-06	5.60E+00	1.43E-06	5.60E+00	1.43E-06	5.60E+00
3a (Scrubbed)	5.85E+03	7.98E-07	4.67E-03	2.66E-06	1.55E-02	3.99E-06	2.33E-02
3a (Unscrubbed)	8.70E+04	1.56E-06	1.36E-01	5.21E-06	4.53E-01	7.82E-06	6.80E-01
3b (Scrubbed)	2.05E+04	7.98E-08	1.63E-03	2.66E-07	5.44E-03	3.99E-07	8.17E-03
3b (Unscrubbed)	3.04E+05	1.56E-07	4.76E-02	5.21E-07	1.59E-01	7.82E-07	2.38E-01
4	n/a	n/a	n/a	n/a	n/a	n/a	n/a
5	n/a	n/a	n/a	n/a	n/a	n/a	n/a
6	n/a	n/a	n/a	n/a	n/a	n/a	n/a
7	4.06E+05	1.24E-04	5.02E+01	1.24E-04	5.02E+01	1.24E-04	5.02E+01
8	1.03E+06	1.78E-05	1.84E+01	1.78E-05	1.84E+01	1.78E-05	1.84E+01
TOTALS:		1.74E-04	74.38	1.74E-04	74.82	1.74E-04	75.14
Increase in Dose Rate					4.4E-01		3.1E-01
Increase in LERF				3.64E-07		2.61E-07	
Increase in $CCFP_{\%}$				0.3		0.2	

the plant. As discussed in Section 2 of this report, the risk acceptance criteria of RG 1.174 is used here to assess the ILRT interval extension.

The 1.81E-7/yr increase in LERF from extending the TMI-1 ILRT frequency from 1-per-10 years to 1-per-15 years falls into Region II ("Small Change" in risk) of the RG 1.174 acceptance guidelines. Per RG 1.174, when the calculated increase in LERF due to the proposed plant change is in the range of 1E-7 to 1E-6 per reactor year, the risk assessment must also reasonably show that the total LERF is less than 1E-5/yr.

As discussed in the TMI-1 PSA documentation, the following TMI-1 PSA release categories contribute to the LERF risk measure:

- RC102
- RC104
- RC202
- RC204
- RC302
- RC304
- RC305
- RC306
- RC402
- RC404
- RC406
- RC408
- RC501
- RC502

Comparison of the above list to the TMI-1 Level 2 internal events PSA results summarized in Table 2-2 of this report, the LERF contribution from internal events is estimated at

3.05E-6/yr. Using the TMI-1 IPEEE results, the LERF contribution from external events is estimated at 2.05E-6/yr (as discussed earlier, External Floods are excluded from consideration as LERF contributors). Therefore, the total LERF for TMI-1 is estimated at 3.05E-6/yr + 2.05E-6/yr = 5.10E-6/yr, which is less than the RG 1.174 limit of 1E-5/yr.